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Research Reactors for the Universities

IN March of last year we cheerfully forecast that by the end of 1960 there would be at least three and possibly four Universities which would be constructing reactors for nuclear engineering research. At the time we believed we were making a reasonable allowance for the ponderous movements of Government departments, but we fear we had not calculated that the mills would grind quite so slowly nor that it would be necessary for the stones to be changed quite so often.

There are currently 10 proposals under consideration, some of which have been completed in great detail for a long time, while others are of more recent origin and some are still in the tentative phases. In some quarters the impression has been given that all requests for financial assistance were made only in the past few months, but at least one grants authority was approached two and a half years ago. Other Government organizations have become involved as time has gone on and the responsibility for making decisions has progressively changed hands. Delay followed delay to the mounting frustration and exasperation of all concerned.

When the Universities first began to develop their atomic energy interests some four or five years ago and to build up departments of nuclear engineering and radiochemistry thoughts turned towards the desirability of building high flux reactors, but it soon became clear that not only would the capital cost of such a project be too high but also running costs and administrative complexities would be on a scale beyond the resources of a single University, or even two Universities combined, such as Manchester and Liverpool.

In 1957 the National Institute for Research in Nuclear Science was established and one of its avowed objects was "to provide, equip and operate, for common use by Universities and by other Institutions and persons engaged in research in nuclear and allied fields, facilities which, by reason of their size or cost or otherwise, howsoever, were beyond the scope of the individual Universities, Institutions or persons aforesaid." The high flux reactor clearly fell into this category but when proposals were examined in greater detail, enthusiasm waned and it became clear that as the active faculties in the Universities were the engineering faculties, requirements could be better met by smaller, low power reactors, such as the Argonaut which had been developed in the United States.

Without reference to a particular design it is difficult to define a power level at which the operation of a reactor changes from being a major plant operation to one that can be accommodated within the framework of normal University life. But in round terms the running of a reactor whose power is 1 MW or above can be considered outside the scope of University departments, whereas for 10 kW and below the problems of fuel handling, general safety and organization are within the capabilities of a department. There is also a significant difference in capital cost between a high and low power reactor. A standard Jason, for example, can be purchased for around £50,000 and a reasonably complete installation with the usual number of modifications will cost in the region of £100,000. Additional site buildings may double this figure but this can be more properly regarded as expansion of the University's general facilities rather than inherent in the reactor cost. For the reactor itself a top figure of £100,000 is a reasonable one to take for the purpose of discussion. In terms of modern installations and major plant which the Universities now expect to have at their disposal this is not an exceptional figure and the Universities were given to understand that sympathetic consideration would be given to requests for funds of this order. It is worth noting that the figure is some two orders of magnitude below that for the proton synchrotron at the Rutherford High Energy Laboratory at Harwell.

Four university departments took the lead; in London, Queen Mary College and Imperial College independently, while Liverpool and Manchester planned a joint venture. Queen Mary College first examined the practicability of purchasing a reactor from the United States and went so far as consulting with manufacturers and determining from the U.K. AEA that there should be no barrier to the import of fuel. It then became clear that their requirements could be better and more conveniently met by a reactor that was being put on the market by a British company (Jason, by Hawker Siddeley) and negotiations were, therefore, started with the company and a schedule for the project drawn up.

In July, 1958, the Universities Grants Committee was informed of the college's wish to proceed with the purchase of a reactor and in the autumn of that year, when plans had been firmly crystallized, the UGC was again approached. An expert was appointed by them to assess the technical aspects of the proposals put forward, but the assessor was not able to undertake the investigation until the spring of the following year. Then, after due consideration of the project a report was submitted giving strong support. The UGC, however, did not react to the

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report until the summer, when the college was informed that there would be some delay in making a decision about funds. From information gleaned from other sources it began to appear that the UGC was not after all the relevant body to apply to for financial aid. As a result the Department of Scientific and Industrial Research and the NIRNS was informed of the negotiations that had taken place and given details of the University's plans. Individuals on the committees concerned were also, of course, aware of what had been going on through informal meetings and private discussions.

Over the same period Imperial College had been evaluating their own requirements and had come to the conclusion that their needs could best be met by a reactor of their own design. The U.K. AEA were approached for their comments on this proposal and replied with a suggestion that the college collaborate with one of the nuclear consortia. Preliminary proposals met with the approval of the AEA and plans went ahead for drawing up detailed estimates of the cost of a Consort, the reactor designed by the college in collaboration with GEC. These were duly sent to the UGC, DSIR, and the NIRNS, as well as to certain individuals whose views were likely to be of importance. Surprisingly no official acknowledgment of the receipt of the documents was received from the organizations, nor was there any reaction to the complete appraisal of the project which was sent to the UGC in May of 1959.

Manchester and Liverpool who had originally examined the desirability of the installation of a high flux reactor had turned their attention to a low flux unit to be built roughly mid-way between the two Universities at Risley and their proposal was submitted to the NIRNS in July of 1959.

It is not really clear how the Institute became involved, but it would rather appear that this was a self-appointment and a carry-over from the original high flux reactor project in which they were quite clearly involved. In view, however, of the fact that there was not now to be a national effort, but independent University projects, it is difficult to see where the Institute fits in, particularly as the Institute had accepted the principle that the Universities would be better served by owning and operating their own reactors. Surely the Institute was designed to provide facilities for collaboration by the Universities when these were requested; it was not formed to direct the policies of individual Universities and, moreover, it was constituted with a much greater emphasis on physics than on engineering, yet it was the engineering departments that were requiring the reactors.

Nevertheless at a meeting of the Institute's Reactor Subcommittee in September 1959, the matter of individual reactors was discussed in great detail and a national plan drawn up. The proposal was that reactors should be allotted to Universities on a regional basis and that a multiplication of these reactors in any one region should be avoided. It was further implied that if there were areas where no plans already existed for the installation of a University reactor such plans should be stimulated. No account was apparently taken of the existence of the firm projects with their background of departmental development aimed at building up lecture and post-graduate courses to use the facilities and, indeed, it seemed that the whole question was treated as something of an academic exercise. Such an exercise, if conducted, say, five years ago, would of course have been useful and Universities would have known where they stood, but it seems strange that, coming so late in the day, the existing projects should have received so little detailed examination. One suspects further that the use to which the reactors should be put was not closely investigated, nor the clear distinction drawn between post-graduate engineering research and training and the physics and radiochemistry requirement for which there was much less demand.

The UGC then made it plain that they could not support the proposals as their funds were limited and in any case first and foremost these were intended for financing building projects and supporting continuing research. The proper source of money for new research equipment was DSIR.

Now that this point was cleared up, although no definite undertaking was given that finance would be forthcoming there were clear implications that the Treasury would make available a limited amount of money for University reactors and it would be the responsibility of DSIR to determine how this should be allocated. The Institute was given the function of investigating the technical needs for reactors in Universities and the University departments concerned were, almost for the first time, given a chance of stating their cases. This at least formalized the Institute's position but we suggest that an unfortunate precedent has been established in that the Institute is more than a technical adviser and a common meeting ground for different University departments, but is in the position of being a national policy maker with control over the activities of individual departments.

Lately, the Treasury has made it known to the DSIR that a special allocation of public funds would not after all be made for reactors and DSIR would be required to assess the needs of the Universities for reactors in competition with the call on their resources for all other types of research equipment. It was only then that the Universities were told that as this had now become a standard request for grants it would be necessary to complete formal applications. So it was that four definite projects came to be re-submitted between March and August of last year. These were the three projects already mentioned, the fourth being from the Scottish universities. Afterwards two further specific proposals were received from Birmingham and Southampton and tentative inquiries have come from Cambridge, Durham, Leeds and Sheffield. The two London colleges discussed it should be noted have not only concluded firm arrangements with manufacturers but have also had site approval from the Ministry of Power and building licences granted.

It now appears that finance will be made available for three reactors but the Institute will be exercising their control over where they should be built. The decision that money for reactors cannot take the form of a special Treasury grant is not unreasonable (although it is most unfortunate that this was not taken two years ago) but it only serves to further emphasize the equivocal position of the Institute. If there is to be a national plan for University reactors, is it not logical to extrapolate this to a national plan for determining what each University should teach and what research it should undertake in nuclear science or indeed in all fields of science and engineering?

Nuclear engineering already suffers in this country from being too inbred with the AEA having been responsible for the early training of so many scientists and engineers. Economic necessities forced us to narrow our approach to the point where it is already not quite comfortable but the Universities should be a source of independent ideas. Even so this is probably less important than the wider implication that the work of University departments is being controlled by a super-national committee with its inevitable restrictions on free expression and originality.

World Digest

Before the start of the meeting of the ENEA Steering Committee held in Paris last November the ENEA director, M. Pierre Huet, and the IAEA director-general, Mr. Sterling Cole, signed the protocol relating to the bringing into force of the collaborative agreement reached between the two organizations in September. With the signatories in this photograph of the event is Professor Leandre Nicolaides (centre), chairman of the Steering Committee. Mr. Sterling Cole later attended the Steering Committee meeting.

International

A manual on nuclear power costing is being prepared by the IAEA to serve as a guide to less developed countries considering the acquisition of nuclear power plants.

Spain has been admitted to membership of CERN. Fourteen countries now belong to this organization. Decision to admit Spain was unanimous.

Dr. Victor F. Weisskopf, formerly with the Massachusetts Institute of Technology, has been appointed director-general of CERN. He will take up his duties in August, when Mr. J. B. Adams, the present director-general, will take up full-time duties as director of the U.K. AEA plasma physics laboratory at Culham.

An international centre for the collection and distribution of physical data and charts of dose distribution related to high energy radiation should be established by the IAEA, it was suggested by a panel of radiation physicists meeting in Vienna recently to review existing data on dose distribution. They also recommended that the Agency published an international catalogue of available distribution charts and some illustrative atlases.

A merger of the executive offices of the three European communities—the Common Market, the Coal and Steel Community, and Euratom—is sought by members of the European Parliamentary Assembly. At present Euratom and the Common Market have their offices in Brussels, while the coal and steel pool administrators and the secretariat of the Assembly are based in Luxembourg. Assembly meetings take place in Strasbourg.

CERN headquarters at Meyrin, Geneva, were recently the scene of an intergovernmental conference on space research attended by representatives from 10 European countries. An agreement was signed for setting up a preparatory commission to study the possibility of establishing an organization similar to CERN.

Details of research contracts placed by Euratom are to be regularly published in the "Journal Officiel des Communautes Europeennes." To date Euratom has concluded 47 contracts—24 of them under the U.S./Euratom research and development programme, seven for the ORGEL project, seven related to studies carried out by the Scientific Data Processing Research Team (GRISA), six contracts of association, two on behalf of the Central Nuclear Measurements' Bureau and one on behalf of the Ispra establishment.

Trade Union experts representing 30 national trade union federations in 16 European countries, the United States and Canada, have been meeting in Dusseldorf to hear about and discuss the effects of the peaceful use of nuclear energy on the interests of workers.

A "Geneva" Proposed for 1963

The first two weeks in August, 1963, are the dates now proposed by the United Nations scientific advisory committee for the next Geneva conference on the peaceful uses of nuclear energy. Earlier it had been suggested that it should be held in 1962. It would have clashed, however, with the 6th World Power Conference, to be held in Australia, and the Foratom Congress in Paris. Reporting the recommendation, Mr. Dag Hammarskjöld said the committee were unanimous in their proposals and reaffirmed their convictions that a third conference of the 1955 and 1958 type was of importance in the exchange of scientific information and the reviewing of developments in the nuclear field. They felt 1963 would "provide the best opportunity for the presentation of important new information in this field." It was also their view that the conference should not be held later. Outside the United Nations doubts have long been expressed as to the value of a mammoth conference, particularly as there is a considerable exchange of information at the multitude of specialized and often large conferences and symposia that are continually being organized by the IAEA and other bodies. Many people, too, have discomfiting memories of the 1958 conference and rebel at the idea of such gargantuanism being revised—there were too many papers, sections and people. It should be ensured right from the start that the next conference will be organized and maintained in manageable human proportions.

United Kingdom

Capacity of the nuclear power station, the CEGB hope to build at Wylfa, Anglesey, might be as much as 800 MW, it was revealed when the CEGB announced in December that construction permission was being sought from the Minister of Power and local planning authority. It is proposed to have the station in commission in 1967 and if built it will be connected to the national grid system at 400,000 volts. No indication was given as to whether the station would be the last of the magnox plants, as has been suggested.

Construction of Nimrod, the 7GeV proton synchrotron at Harwell will probably not be completed until July, 1962, according to the third annual report of the National Institute for Research in Nuclear Science. This is six months later than the previous estimate. Reason given for the revised completion date is that as the project has progressed it has been possible to make a more accurate prediction. Work on the accelerator is well in hand. During the year financial support was received from the D.S.I.R. for one of the University bubble chamber projects and a start was made on the buildings needed to house the project. An initial study has been carried out on the second bubble chamber project and following an assessment of the proposed design, the NIRNS have agreed to provide financial support subject to Treasury approval. Expenditure by the Institute during the 1959-60 year was approximately £5.4 million.

Finance for three reactors to be built by the Universities is to be made available shortly by D.S.I.R.

A decision is expected very shortly on the application by Liverpool and Manchester universities to the Minister of Power for a nuclear site licence in respect of their proposed reactor project at Risley. When financial arrangements have been made they plan to install a Jason.

The Reactor School, Harwell, is to hold its fourth advanced course on "The Principles of Radiation Protection" from April 19 to July 18.

No decision has been made by the AEA yet as to whether to construct a prototype steam generating heavy water reactor (SGHWR). It is understood, however, that priorities have been reduced and enthusiasm for the project is diminishing.

A 400,000-volt supergrid system is to be adopted by the CEGB. The purpose is to reduce the number of high-voltage transmission lines needed to meet the country's increasing power demands. The majority of existing 275,000-volt supergrid lines were originally designed for possible conversion to 380,000 volts. Where increased transmission capacity is required these will be reinsulated and used for the new high voltage. For new main trunk lines the Board plan to introduce a new design of tower carrying four conductors per phase, as compared with two per phase on the 275,000-volt lines.

"Heat Transfer II" is the title of a series of 10 lectures to be given at the Northampton College of Advanced Technology, London, for students of graduate and H.N.C. standard. The lectures will be given on Tuesday evenings from 7 p.m. onwards, starting January 10. Also starting that date will be 10 lectures on the "Fatigue of Metals and Other Materials."

Aspects of DFR Discussed

AT the Institution of Mechanical Engineers' one-day symposium on the Dounreay Fast Reactor (which we shall be reporting in greater detail next month) an account was given of the commissioning procedure and problems, in addition to specific design and constructional aspects of the plant itself. The conclusion was drawn that although the commissioning period was lengthy, due in part to the complexity of the system and the necessity for repeating any modifications to one primary heat exchanger, in all 24, problems were no worse than those to be expected on a new processing plant. After being shut down for modifications the reactor has now been operating for two months and is at present working at a power of about 100 kW. It is hoped to raise this to 5 MW at the end of January when the plant will be shut down again and further modifications made to the control elements. It is probable also that a new core of uranium-molybdenum alloy will replace the present uranium-chromium alloy and the reactor will be made critical in the early summer and should be operating at 20 MW by the end of the year.

Australia

New engineering and metallurgy laboratories have been opened at the Lucas Heights establishment of the AAEC.

MOATA, Australia's second reactor, is expected to go critical towards the end of February. Shipment of the components from the suppliers, Advanced Technology Laboratories, of California, has been completed and assembly of the reactor at Lucas Heights, near Sydney, started during November. Details of this reactor, which will cost about £A65,000, were given in *Nuclear Engineering*, October, 1960.

About £A100,000 is to be spent on equipment for studies in advanced plasma physics at Sydney University. Plans for this work have been drawn up by Professor C. N. Watson Munro, professor of thermonuclear physics at the university, following an eight-month tour of the fusion research centres throughout the world. It is believed that half of the cost of the equipment has already been contributed by an industrial company.

Buildings for the Australian Atomic Energy Commission's Nuclear Science Institute and canteen at Lucas Heights have been completed. The Institute, with its projecting lecture hall, is on the left of this illustration. Contractors were Taylor Woodrow (Australia).

A gift of $250,000 to aid scientific research has been made to Sydney University by an American businessman and his wife, Mr. and Mrs. Cecil H. Green, of Dallas, Texas. The money will pay for a new electronic computer to be used for work in nuclear physics.

Brazil

President Juscelino Kubitschek and other high-ranking Brazilian officials attended the inauguration of the Triga reactor at Minas Gerais University. The reactor went critical on November 6 and it is the first Triga to be in operation in South America.

Canada

A uranium-steel alloy has been prepared by scientists at the Ministry of Mines. A patent for the process has been taken out and is available to the steel industry. According to the Minister, Mr. Paul Comtais, the strength of the steel, its corrosion resistance and other properties at high temperatures were improved by the addition of uranium. The proportions used were a pound of uranium to a ton of steel.

Colombia

Hawker Siddeley Nuclear Power Company have been negotiating with the Instituto de Asuntos Nucleares for the installation of a Jason reactor and a high energy cobalt-60 source. The company believe the country has scope for the development of a programme based on the medical and agricultural applications of radiation and radioisotopes for which the facilities they offer would be ideally suited. A decision is expected following the visit of an IAEA study group to the country early in the year. It is understood that other countries have been similarly approached by the company. Hawker Siddeley entered the cobalt-60 radiation field with their acquisition of the irradiation equipment manufacturers, Nuclear Engineering.

Egypt

The German Federal Republic have offered to assist the United Arab Republic in the development of nuclear energy for peaceful purposes. Preliminary talks have been held between Egyptian officials and the Federal Ministry for Atomic Affairs, Bonn. A firm agreement is expected very shortly.

Guatemala

A Cobalt-60 source has been donated to the Nuclear Energy Commission by the U.S. AEC for use in the teletherapy department of the Gautemalan Cancer Institute.

West Germany

The 15 MW BWR plant at Kahl, near Frankfurt, which recently went critical, is expected to be operated for two or three months at power levels of up to 1 MW for testing purposes. It will then probably be run at about 50% of its rated capacity for three months before being taken up to full power. The plant is said to have cost nearly DM40 million (about £3¼ million) to build. The main contractors were Allgemeine Elektrizitats-Gesellschaft (AEG), who built it to General Electric of America designs for Rheinisch-Westfaelisches Elektrizitatswerk (RWE) and Bayernwerk.

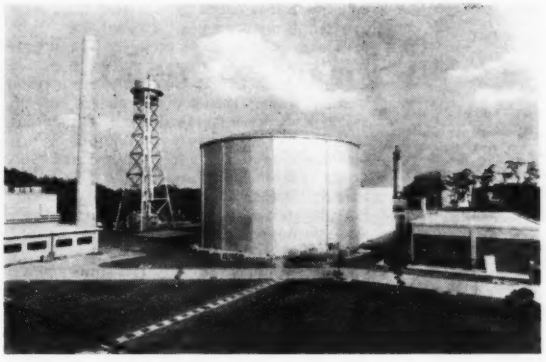
India

Radioisotopes and Radiation in Entomology were the subjects of a symposium held in Trombay recently at which experts from nine countries presented over 25 papers. The symposium was arranged by the IAEA in conjunction with the Indian Government.

Iran

An IAEA expert has arrived to supervise the installation of an A.M.F. 5-MW pool-type reactor at the University of Teheran's nuclear centre being developed at Amirabad, a few miles from the capital. The U.S. Government is supplying $350,000 towards the project.

Japan



The Atomic Energy Research Institute at Tokai Mura, with the 10MW(t) research and testing heavy water moderated reactor built by A.M.F. in the foreground. The Japanese government recently received from the U.S. a cheque for $350,000 to go towards the cost of this reactor.

South Africa

No changes are believed to have been made in existing arrangements for the sale of uranium to the U.K. and U.S.A. as a result of the recent discussions held between the Atomic Energy Board and the Combined Development Agency. It is understood that the board had hoped that existing contracts could be "stretched out." Under the present arrangements they are due for completion between 1963 and 1966.

Potchefstroom University, in the Transvaal, have bought from the Cavendish Laboratory, Cambridge, their remaining Cockcroft-Walton accelerator. A 2MeV machine, it was sold to the university for only £4,000. The laboratory is in the process of turning over from nuclear to solid state physics as its area of work and the only remaining large piece of nuclear physics equipment is a 5MeV Van der Graaff. About two years ago another South African university bought a 1MeV machine from them.

Sudan

Preparations are well advanced for the establishment of a nuclear energy institution by the Ministry of Works, reported President Abboud in a nationwide speech on November 17.

Switzerland

The three nuclear groups, Suisatom, Enusa, and Thermatom (Consortium), have jointly formed Compagnie Nationale, for the construction at Lucerne of a heavy water pressurized tube reactor using natural uranium. The Federal Government will provide Sw. Fr. 70 million for the project, which is based on the proposals previously prepared by Thermatom for a 30 MW reactor. Enusa previously had plans for the contruction of a 5 MW BWR and Suisatom for a 27 MW BWR from America.

U.S.A.

An operating licence is to be issued by the AEC to Wisconsin University for a 10 kW pool type reactor. Supplied by General Electric, it is estimated to cost $285,000, $150,000 of which is being contributed by the AEC. The rest of the cost will be borne by the university. Fuel will be loaned free of charge.

With the objective of building a 300-500 MW power plant in New York State, Empire State Atomic Development Associates, a consortium of seven utility companies, are contributing to a three year development programme to be carried out by General Atomic. They are allotting $4.5 million to the project, while General Atomic's share will be $4.2 million. The latter figure is in addition to the funds already spent by General Atomic on the HTGR concept.

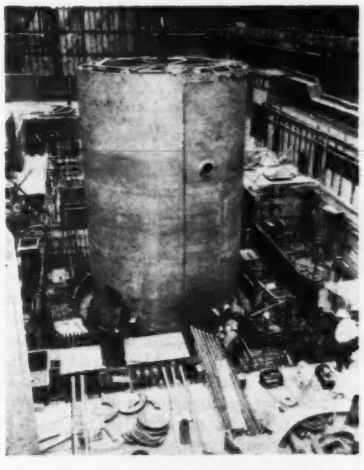
An extension of seven months has been given to the Power Development Company by the AEC for the completion of the Enrico Fast Breeder reactor at Lagoona Beach. The company had asked for their construction licence to be extended 12 months, to December 16, 1961; an extension, however, has been granted for only up to July 16, 1961. Work has been delayed on the project due to the setting aside of the original licence on safety grounds by a panel of the U.S. Court of Appeals for the District of Columbia in June, 1960. The AEC and PRDC have appealed against the decision and are now awaiting further developments from the Court of Appeals and the U.S. Supreme Court. The AEC emphasize that the extension of the construction permit allowed to PRDC is subject to the findings of these judiciary bodies.

Pin-type fuel elements replace the plate-type elements originally used in the ML-1, the 2MW(t) gas-cooled reactor experiment at Idaho Falls, leading to a reduction in the size and weight of the core. The object of the experiment is to provide data for the development of portable plants for military use in remote locations (the designation ML-1 stands for "Mobile Low Power Plant No. 1") and the new core, successfully tested on Nov. 23, 1960, brings the objectives nearer to feasibility. The core contains 55 elements each composed of 19 pins in two concentric rings about a central pin. The pins are 0.25 in dia. Hastelloy-X tubes loaded with highly enriched uranium oxide in the inner ring and a mixture of sintered uranium oxide and beryllium oxide pellets in the outer ring. The entire core is 22 in dia. and 28 in long. ML-1, designed to have an electrical output of between 300 and 500 kW is scheduled for test operation during the coming summer.

European Ship Proposed

CLOSE on the heels of the IAEA conference on ship propulsion, held from November 14-18 at Taormina, Sicily, came the announcement that the ENEA is to establish a study group for exploring the opportunities and possibilities for European co-operation in this field. At the conference Mr. Boxer, the ENEA representative, suggested the construction by ENEA countries with maritime interests of a jointly owned and operated nuclear propelled ship. It would lead to a reduction in the research and development costs in individual countries, he said, providing for them a short cut in the same way as other joint European projects. It would also facilitate the establishment of a common policy for the acceptance of nuclear ships in European ports. It will be interesting to see whether these proposals are reflected in the deliberations and action of the new study group. Mr. Boxer gave his views in one of the 40 papers presented at the conference, which was attended by 175 delegates representing 19 countries and six international organizations. What else was said and discussed can be found on page 33 of this issue, where a report of the proceedings is given.

A public inquiry is being held in St. Paul, Minnesota, to consider whether authorization should be given for the operation of the 22 MW BWR power plant under construction at Elk River.



The 65 ton reactor vessel for the 75MW sodium cooled reactor plant at Hallam, Nebraska, is lowered into place.

Structural steel for the magnet building of the zero gradient proton synchrotron at Argonne goes up. Foundations for the 5,000-ton ring of magnets are already in place. Scheduled completion date for the project is 1962.

A novel scheme of pressure suppression containment has been proposed for the BWR plant to be built at Humboldt Bay, California, for the Pacific Gas and Electric Company. The basic components of the system will consist of a pressure-tight, carbon-steel vessel called the dry well which will enclose the reactor vessel and associated piping up to the first valve in each line and the control-rod drives. The dry well will be surrounded by another pressure container constructed of reinforced concrete and lined with thin steel plate. This container, known as the suppression chamber, will be partially filled with water, referred to as the suppression pool. The dry well and suppression pool are designed to function in the event of any failure occurring to reactor pressure vessel fittings or primary system piping up to the first valve. Steam or saturated water from the break would flash to steam and fill the dry well, from which it would be vented to the suppression pool where it would condense. The reactor and containment systems will be located underground, enclosed in reinforced concrete. The plant, which will have an output of between 50 and 60 MW, has been designed by General Electric. Construction is scheduled to be completed in March, 1962, and initial criticality reached by June, 1962.

GRANTS totalling $1,547,538 *have been approved by the AEC for buying laboratory equipment needed to initiate nuclear education programmes in* 113 *teaching establishments. They bring the total assistance provided during the past four years by the Commission in this way to* $18,407,892. *The equipment grants just awarded are within three general categories : physical sciences and engineering awards to* 59 *institutions, worth* $1,040,174 ; *life science awards to* 50 *institutions, worth* $423,255 ; *and isotope technology awards to* 10 *institutions worth* $84,109.

The plutonium recycle test reactor at the AEC's Hanford plant near Richland, Washington, is undergoing start-up tests with various patterns of fuel loadings. Purpose of this reactor, which is a heavy water moderated, cooled pressure-tube type, is to test plutonium-bearing fuel elements for the purposes of developing plutonium as a fuel for industrial power. Design power output of the unit is 70 MW(t). No electricity generating facilities are included. The reactor is operated for AEC by General Electric.

AEC Switch Contractors

SECOND THOUGHTS over the tenders submitted for the 50-MW BWR plant to be built for use by the Los Angeles-Pasadena utility companies have led the AEC to change their choice of bid for further negotiation. Tenders were invited in December, 1959, and were received by the following May. A month later the AEC indicated that the $12.5 million proposals of General Electric had been accepted. Last November, however, they announced that they had decided, instead, to negotiate with the Allis-Chalmers Manufacturing Company. According to early reports on the changeover, GE's higher bid had been accepted because their design included an internal steam separation system which the AEC considered as a major advance in BWR technology. Soon after its acceptance General Electric told the Commission that although it had previously been thought necessary to " prove out " the internal steam separation technique in a prototype reactor plant, which is, in fact, what the Los Angeles-Pasadena station will be, it could be put into use in a 300-MW plant without such a test. As a result AEC felt it was not such an important consideration in the L.A.-P. plant and reviewed the matter. According to the Industrial Forum, however, the AEC restudied the tenders following protests from the unsuccessful bidders. It was then found that an error had been made in evaluating the Allis-Chalmers bid which made it $500,000 higher than it actually was. Their price for the nuclear portion of the plant is $10,760,370 and this, according to the Forum, includes internal steam separation. The plant will be operated for at least five years in order to provide data for the design and construction of a 300-MW station. Scheduled completion date for ICBWR (Improved Cycle Boiling Water Reactor), as the prototype is designated, is early 1964.

Reorganization of the AEC's licensing and regulation division has taken place. Some of the former branches have been split up into smaller groups and three new assistant directorships created to take responsibility for these groups. The licensing branch has been divided into the facilities branch and the nuclear materials branch under the general charge of an assistant director for facilities and materials licensing. The hazards evaluation branch has been split up into the research and power reactor safety branch, test and power reactor safety branch, special projects branch, criticality evaluation branch, and site-environmental branch. They are the responsibility of the assistant director of nuclear facilities safety. The third newly created assistant directorship has responsibility for nuclear materials safety.

Thirty-five ton payloads could be launched into space by means of nuclear propelled rockets by 1967, it was predicated by Mr. John W. Simpson, vice-president of the Westinghouse nuclear power division, at a recent meeting of the Pennsylvanian branch of the American Society of Nuclear Engineers.

From March 31, 1962, onwards purchases of uranium concentrates by the AEC will be charged at a standard rate of $8/lb of U_3O_8. At present the AEC have 27 contracts for the supply of uranium concentrates and prices paid under these contracts for deliveries before March, 1962, range from $7.72/lb to $9.9/lb. Exceptions to the new standard rate will be made where contracts previously ending in March, 1962, have been extended. Some of the contracts cover delivery periods up to December, 1966, and their total value is estimated at $1,552,800,000. The aggregate quantity of U_3O_8 to be delivered under these contracts is given as about 190,674,000 lb, of which 74,307,00 lb is scheduled for delivery before 1962.

Venezuela

A cheque for $350,000 was presented to the Venezuelan Government by the U.S. AEC at a ceremony on November 22 to mark the inauguration of the country's first reactor—a 3-MW pool-type unit constructed by General Electric installed at the Institute of Scientific Research, Caracas. The money was paid under the terms of the " Atoms for Peace " offer made by President Eisenhower in 1955 for assisting countries in operating training and research reactors.

Uranium oxide has been found to be present in phosphate mineral deposits in the Andes, according to the Ministry of Mines. Surveys have recently been carried out in the Andean districts to determine mineral resources for use in the manufacture of fertilizers. The uranium oxide occurs at the rate of about 10 gram to every ton of mineral.

Yugoslavia

An agreement has been signed between the Yugoslav Federal Commission for Nuclear Energy and the Italian National Committee for Nuclear Energy for a programme of technical collaboration. It covers basic and applied research, studies on raw materials, the application of radioisotopes and radiation hygiene. There will also be an exchange of staff between the two organizations.

SIZEWELL

Contract Placed for Seventh "Commercial" Station

SIZEWELL, seventh of the "commercial" stations, will be situated on the Suffolk coast, a few miles south of Dunwich, between Southwold and Aldeburgh, and is to be built by the English Electric—Babcock and Wilcox—Taylor Woodrow Atomic Power Group. Site work is scheduled to begin in April 1961, and it is expected that the station will be operational in 1966.

Rationalized Design

The efforts of the designers towards compactness and simplicity have resulted in several novel features. Both reactors are housed in a common building, so that pile cap services (cranes and charging machines), interconnecting services, control and change rooms can be integrated into one unit, instead of practically constituting two distinct stations, and a very considerable amount of duplication is avoided.

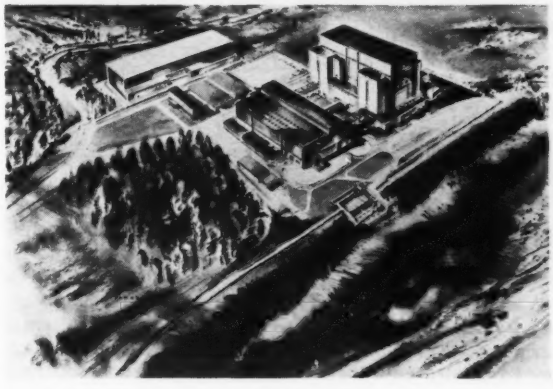
Simplicity is also the keynote of the steam side. Although the guaranteed output is 580 MW, the number of turbines has been reduced to two, and the heat exchangers to eight for the whole station, as compared with the 12 heat exchangers and six turbines for Hinkley Point, the Group's previous station. The use of only two turbines has resulted in a considerable reduction in the size of the turbine hall. A further simplification is the use of constant-speed blowers, coolant control being by a combination of vane control on the blowers and gas by-passing.

Advances made in the design of the heat exchangers, or boilers, have enabled the steaming capacities to be increased by nearly 50% over those at Hinkley Point, with very little increase in shell size (90 ft-91 ft 6 in high, and 21 ft 6 in-22 ft 6 in diameter). A low-alloy high tensile steel is to be used for the shell. New ground is broken in the tube bank design; the H.P. and L.P. superheater banks being placed side by side, so that the final steam temperatures are identical in both systems. The use of internal trifurcating connections from three tube elements to a single tube and thermal sleeve will considerably reduce the number of shell penetrations.

The by-pass valve controlling the gas exit temperature is mounted within the shell, the drive being taken through the shell via a seal of the displaced-bellows type.

Economy

The accompanying table gives some of the general parameters of the station. Unfortunately, many of the more interesting figures are, for reasons of commercial security, not at present available—with propriety—for publication. It is, therefore, not possible to give anything like a complete comparison with previous designs, but it is possible, on the published information, to draw certain conclusions. It would appear, for example, that overall optimization has been the keynote of the design; that the designers have not gone all out for thermal efficiency is shown by the figure of 30.5% which, while high, is still 2% less than the published figure for Dungeness. Thermal efficiency is not, however, always synonymous with minimum overall cost, and might well be outweighed by the saving in interest charges on reduced capital cost. Some of the steps taken in this direction have already received brief mention; it is reasonable to assume that further savings will be made in the turbine house—not only in the building and ancillary plant, but in the turbines themselves. These, with a capacity of 325 MW each are, when considered by the standards of only a few years ago, enormous machines. It would appear within the bounds of possibility that they might be based on what is often referred to as "CEGB" design derated for the lower steam pressures and modified for mixed flow steam conditions (the exhaust blading is said to be the longest ever manufactured). If so, this would account for further economies; the cost of developing sets of this size *ab initio* can well be imagined.



Artist's impression of station.

GENERAL PARAMETERS

Station	
Net electrical output	580 MW
Net thermal efficiency	30.5%
Reactors	
Reactor heat rating	950 MW(t)
Pressure vessel diameter	63 ft 6 in
Pressure vessel thickness	4.175 in
Reactor inlet pressure	279 p.s.i.a.
Inlet temperature	N.A.
Outlet temperature	N.A.
Fuel rod diameter	1.1 in
Fuel charge/reactor	321 tonnes
Mean fuel heat rating	2.95 MW(t)
Number of elements per channel	7
Number of channels	N.A.
Weight of graphite	N.A.
Mass gas flow	N.A.
Steam Plant	
Number of heat exchangers	8
H.P. drum pressure	720 p.s.i.a.
H.P. superheater outlet temperature	391°C (737°F)
L.P. drum pressure	305 p.s.i.a.
L.P. superheater outlet temperature	390°C (735°F)
Steam pressure at H.P. boiler stop valve	677 p.s.i.
Steam temperature at H.P. boiler stop valve	389°C (734°F)
Steam pressure at L.P. boiler stop valve	268 p.s.i.
Steam temp. at L.P. boiler stop valve	389°C (734°F)
Vacuum	29 in Hg
H.P. steam flow from each boiler	5.4×10^5 lb/h
L.P. steam flow from each boiler	1.9×10^5 lb/h
Shell dimensions	22 ft 6 in dia. × 91 ft 6 in high
Number of main turbines	2
Output of each turbine	324.75 MW
Main Circulators	
Number of circulators	8
Type of circulators	Vertical, single stage axial with variable stagger inlet guide vanes
Drive	Vertical squirrel cage induction motor
Motor H.P.	N.A.
Civil Engineering	
Height of reactor building	193 ft
Length of reactor building	400 ft approx.
Mean thickness of concrete main shield	5 ft 6 in
Mean thickness of concrete supplementary shield	5 ft
Distance across flats of main shield	72 ft 3 in
Height of concrete main shield	86 ft 2 in
Thickness of pile cap	11 ft 9 in
Foundation of reactor building	Reinforced concrete raft 225 ft × 110 ft × 8 ft thick
Turbine hall overall dimensions	Approx. 385 ft × 155 ft wide × 94 ft high
Pumphouse dimensions	125 ft wide × 133 ft long

Thin Tube Plates for H.P. Heat Exchangers*

by D. E. HARTLEY, M.A.
(Queen Mary College, Nuclear Engineering Laboratory, London)

In Part 1[1] a theoretical analysis showed that the use of very thin tube plates (16 s.w.g.) in heat exchangers operating at 1,000 p.s.i. might be possible although stresses appeared extremely sensitive to small temperature changes. Part 2 describes some of the experimental work undertaken to test the theory together with preliminary results on the straining of thin metal plates. The experiments indicate that thin tube plates are stronger than the theory predicts.

Part 2. Experimental Work

THE experiments performed were not strictly systematic and do not form a clearly defined entity; this was partly as a result of the way in which the problem of thin tube plates arose, partly because the tests gave some unexpected results, and partly because of the limited time which was available for the project. Destruction tests were made on two small heat exchangers; some small strips of metal (simulating the outer edge of a tube plate) were deflected until they broke; similar thin strips of metal were fatigue tested in bending, to deflections which theoretically would lead to fracture; finally, an exchanger was subjected to severe pressure cycling until failure occurred.

Destruction Test on 22 s.w.g. M.S. Tube Plate

The specific need for thin tube plates at high pressures arose in a project where a compact heat exchanger having ¼-in OD 30 s.w.g. tubes was estimated to need tube plates 3¼ in thick. The first unit based on that design comprised 159 mild-steel tubes ¼ in OD by 30 s.w.g., 13½ in long (triangular tube pitch of 0.35 in) contained in a 5-in bore

Fig. 1.—Unit No. 1; end caps removed.

¼-in wall mild-steel shell; the tube plates were 0.028 in thick mild steel. Ordinary wide-face joints were employed with fibre gaskets and the tubes were fusion welded to the tube sheets which had previously been "nippled." Fig. 1 shows the unit without headers. The pipe at the centre of the shell was for attaching a glass tube for observing changes in water level (the shell being filled with water during the test) as an indication of the beginning of leakages. Arrangements were made to pressurize the tube side of the exchanger through a central connection which can be seen in Fig. 2. Due to difficulties encountered in the welding 14 tubes were blanked off after they had first been filled with solid bars.

Hydraulic pressure was applied to the tube side of the unit (the shell side being full of water at room pressure), and the pressure raised to 1,000 p.s.i. and maintained for 20 min. As no sign of leakage was apparent, the pressure was then increased in steps of 50 p.s.i. to 1,450 p.s.i., at which point leakages occurred.

Fig. 2.—Unit No. 1; complete assembly.

Examination of the tube bundle after it had been cut out of the shell, showed that the tube plates were not fractured but that several tubes in the outer row had buckled and cracked (see Fig. 3). The tube plates had deformed in the region of the buckled tubes.

Referring now to the theory and using the same nomenclature as before:

The flexibility ϕ (equn. 7, Part 1), is 0.78, X is 2.36, s_o = 0.30 in and s=0.163 in

the outer edge deflection δ_o is $7.6 \times 10^{-4}\ w_t$ in,

and the total stress $84.2\ w_t$.

From these figures the stress-pressure relationships in Table 1 are deduced which show that the plate was not fractured at a pressure 2.4 times greater than that estimated to cause failure.

Estimates from Ref. 2 show the exchanger to be suitable

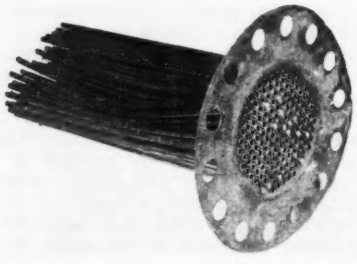


Fig. 3.—Unit No. 1; tube plates 0.028 in thick. Failure of tubes at 1,450 p.s.i.

* The work described in these papers was done when the author was with The Superheater Company, Limited, and Heat Exchangers, Ltd. The views expressed are not necessarily the official views of the Superheater Company.

Table I. Estimated Tube Plate Stresses

Pressure, p.s.i.	Stress, ton/in²	Remarks
160	6	Design.
597	22.5 (measured UTS)	Estimated fracture.
1,450	54.5	Tubes failed in test, plates locally bent.

for operation at 70 p.s.i., an expected failure at 262 p.s.i., and at 1,450 p.s.i. a stress in the tube plate of 124 ton/in². The tube plate thus withstands at least 5.5 times the pressure estimated to cause failure according to BSS 1500 (1958).

The buckling estimates for pin-jointed tubes give a value of 1,270 p.s.i.

Destruction Test on 20 s.w.g. SS Tube Plates

For the second test a new tube bundle was designed and constructed comprising 38 stainless steel (18/8) $\frac{1}{2}$-in OD by 18 s.w.g. tubes with stainless steel tube plates 0.036 in thick. The same headers and shell were employed as in the first unit. The tube nest geometry is given in Table 2.

TABLE 2
Tube Arrangement of the Second Unit

Row ..	1	2	3	4
PCD, in ..	$4\frac{1}{8}$	$2\frac{3}{4}$	$1\frac{3}{8}$	0
n	19	12	6	1

In the second unit the aims were to increase the buckling resistance of the tubes and to study the welding problems with stainless steel. These last were resolved by adopting the argon arc process.

Hydraulic pressure was applied gradually to the tube side of the unit and the volume changes on the shell side recorded by means of a calibrated glass scale. Failure occurred in the tube plate due to the tubes buckling at a pressure of 3,200 p.s.i. Figs. 4 and 5 show the tube bundle after the test.

The thin plate constants are:

$s=0.416$ in $s_o=0.307$ in,

$\phi=21.4$, and $X=23.4$.

The estimated deflection is $1.9 \times 10^{-6}\ w_t$

and the estimated total stress is $108\ w_t$.

Some values are given in Table 3.

Thus the tube plate had not failed at a pressure 4.4 times that estimated to lead to failure, whilst calculations made on the basis of BSS 1500 give an operating pressure of 121 p.s.i. and a failure pressure of 484 p.s.i.; at 3,200 p.s.i. the estimated maximum stress is 232 ton/in². The tube plate is thus at least 6.6 times stronger than the BSS estimate.

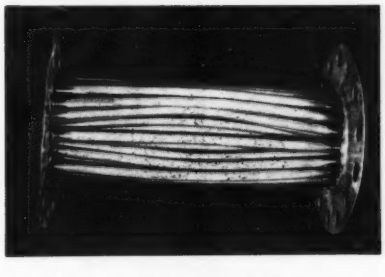


Fig. 4.—Unit No. 2; tube plates 0.036 in thick. Tubes buckled at 3,200 p.s.i.

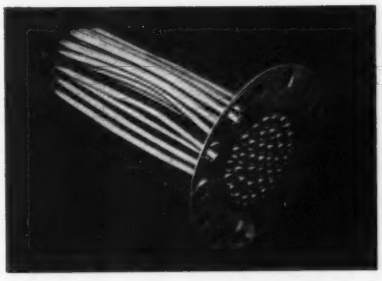


Fig. 5.—Unit No. 2; another view of the tube bundle after test.

The calculated buckling pressure is 11,000 p.s.i., which is considerably in excess of the test result. Clearly the design of exchangers with thin plates will be very much dependent on the resistance of the tubes to buckling in cases where the high pressure is on the tube side of the unit.

Bending Experiments on Thin Sheet Metal

Static deflection tests were made on four types (i.e., geometries) of thin metal strip. The aim was to simulate in a manageable form the deflections at the outer edges of thin tube plates where the theory predicts very high stresses for very small deflections. A rig was designed and this is shown schematically in Fig. 6. The specimen A is a $\frac{1}{2}$-in wide piece of 18 s.w.g. mild steel. Over its central part it is rigidly held by two pieces B of $\frac{1}{8}$-in plate; at its outer edges the specimen is welded to the part C which is 1 in wide. Component C is held by studs to a circular plate D which is supported by (but not attached to) the cylinder E. Loads were applied to the specimen from a Denison Testing Machine via the spindle G. A dial gauge F served to measure the vertical movement of the centre of the specimen to 0.0001 in.

Two of the test specimen types were as shown in Fig. 6 with effective beam lengths s of $\frac{3}{4}$ in and $1\frac{1}{4}$ in. The other specimens were more complex; one incorporated a simple expansion joint, the second was designed to give a large span for deflection whilst being restrained against high pressure. As the test results showed the simple specimens to be very flexible, the tests on the two special ones will not be described.

Three specimens of Type 1 were tested ($s=0.75$ in) and

NOMENCLATURE

E	Young's modulus
f	Stress
h	Plate thickness
n	Tube number in a given row
K_h	Deflection per unit load for an element of plate, equation (5), ref. 1
s	Effective span of plate, or of element of tube plate.
s_o	Effective span of outermost strip of tube plate
V	Load on a plate element of breadth z
w	Pressure; w_t is pressure on the tube-side of an exchanger
X	Deflection ratio, equation (16), ref. 1
z	Breadth of an element of plate
δ_o	Deflection of tube plate in outermost strip
ϕ	Flexibility ratio of tube plate element to tube, equn. (7), ref. 1

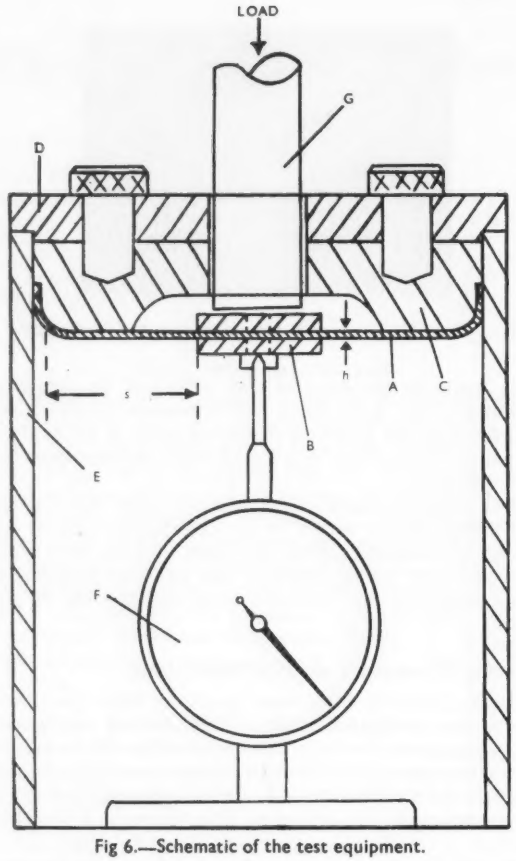


Fig 6.—Schematic of the test equipment.

two specimens of Type 2 (s=1.25 in). Tables 4 and 5 give typical test results, where the loads correspond to $2V$ in Figs. 7 and 8, which are the load deflection curves for the two types of specimen.

The three specimens of Type 1 give almost identical curves up to loads near the breaking points; they break at loads of 0.59, 0.607 and 0.66 ton at deflections of 0.46 in, 0.53 in and 0.53 in, respectively. The two specimens of Type 2 do not agree so well in their test results. One of them failed at 0.61 ton, 1.11 in, the other at 0.70 ton, 1.125 in.

In all cases bending took place locally at the " ends " of the beams (plastic hinges) so that over nearly the whole of the regime the specimens were under tensile loads rather than bending loads; most failures occurred by necking in the centre of one or other side of the specimen with a fracture exactly like that which occurs under direct tensile loads (see Fig. 9).

These measured deflections and loads are very different from those estimated from elastic beam theory. From ordinary beam theory, with both ends encastred, the stresses and loads for given deflections are:—

$$f = 3 E h \delta/s^2 \quad \text{.....(1)}$$

$$\text{and } V = E z h^3 \delta/s^3 \quad \text{.....(2)}$$

Table 3. Estimated Tube Plate Stresses

Pressure, p.s.i.	Stress, ton/in²	Remarks
182	8.75	Design.
725	35	Estimated fracture.
3,200	154	Tubes buckled.

The measured UTS of the steel was 20.3 ton/in² and using this value in equations 1 and 2 leads to estimated deflections and loads for failure as shown in Table 6.

Table 4. Static Deflection Test on Type I Specimen

Load, ton ..	0	0.104	0.44	0.46	0.48	0.49	0.50
Deflection, in	0	0.077	0.285	0.293	0.309	0.317	0.328
Load, ton ..	0.52	0.54	0.55	0.56	0.57	0.58	0.59
Deflection, in	0.336	0.360	0.369	0.378	0.388	0.397	0.407
Load, ton ..	0.60	0.60	0.60	0.60	0.60	0.60	0.607
Deflection, in	0.427	0.435	0.445	0.465	0.475	0.485	0.53 (break)

Thus the loads to cause failure are of the order of 40 times those estimated, the deflections of the order of 70 times those estimated and the flexibility of the plates of the same order as those estimated. The comparison between the ordinary beam and the specimens of the tests is, of course, helped favourably by the curvature of the specimens at their outer edges. However the specimen shapes were based on a proposed design, so that whilst the pure research aspect is not straightforward, the practical significance remains.

Deflection Fatigue Tests

Whilst the static deflection tests just described are very encouraging, the question of fatigue then arises, i.e., if a tube plate is designed in the plastic regime, how often can it be strained a given amount and restored without failure?

To study this problem a rig was designed in which specimens could be oscillated with fixed amplitudes of deflection with a ¼ h.p. motor employed to operate the oscillating rig at 5 rev/min. Two tests only were made.

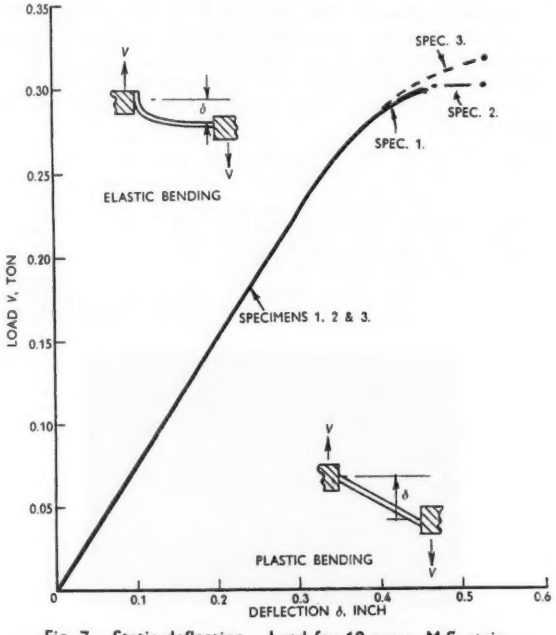


Fig. 7.—Static deflection. Load for 18 s.w.g. M.S. strip, 0.5 in wide ; 0.75 in beam length.

Table 5. Static Deflection Test on Type 2 Specimen

Load, ton ..	0	0.04	0.08	0.10	0.14	0.16	0.18
Deflection, in	0	0.044	0.088	0.1052	0.1481	0.1710	0.1948
Load, ton ..	0.20	0.24	0.26	0.28	0.30	0.34	0.36
Deflection, in	0.2131	0.2535	0.2760	0.2950	0.3152	0.3583	0.3830
Load, ton ..	0.38	0.42	0.44	0.46	0.48	0.50	0.70
Deflection, in	0.3965	0.4436	0.4678	0.4907	0.5182	0.5316	1.125 (break)

The specimens were generally as specimen 1 of the static deflection tests; both had lengths of 0.75 in and were oscillated from zero to 0.01 in; one specimen was of 18 s.w.g., the other of 22 s.w.g. mild steel. In the two cases the stresses, by elastic theory, were 34 ton/in^2 and 20 ton/in^2.

After 10,000 cycles there was no failure, no visible surface cracking and no signs of deterioration in the metal structure under micro-examination.

Fatigue Tests on a Complete Exchanger

An exchanger similar to the second unit but with 14 s.w.g. tubes and a device to stiffen them against buckling was pressure cycled between 200 p.s.i. and 3,200 p.s.i. at a rate of 8 to 9 cycles per minute.

Failure occurred by a weld cracking after 4,500 cycles; but no special conclusions can be drawn from the test. The result is, however, included in support of the fatigue tests on small samples.

Plastic Design of Thin Tube Plates

The tests on the complete exchangers indicate the tube plates to be more than 2.4 times (unit 1), and 4.4 times (unit 2) stronger against pressure loads than the analysis in Part 1 would have one believe. The static tests in deflecting specimens of thin metal show that failures due to bending occur at about 70 times the estimated deflections. Clearly the analysis based on the elastic bending of beams does not give an adequate representation of the strength of thin tube plates and it is far too pessimistic.

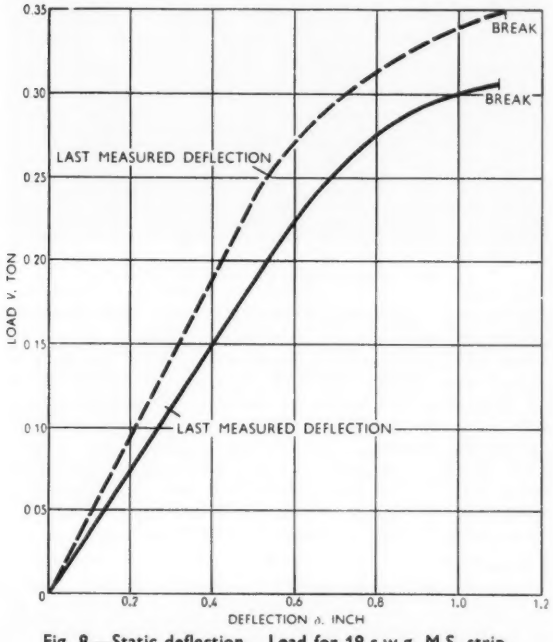


Fig. 8.—Static deflection. Load for 18 s.w.g. M.S. strip, 0.5 in wide; 1.25 in beam length.

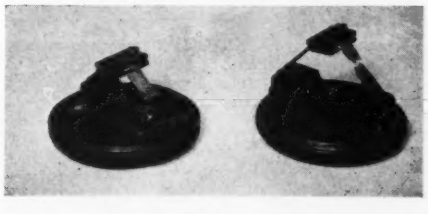
Specific evidence of the strength of thin metal can be found in the literature on bursting discs[3, 4] and from simple estimates based on the characteristics of certain types of paper clips. In the civil engineering field, steel-framed buildings are now frequently designed on plastic strain theory and this is accepted by the British Standards Institution[5, 6].

Table 6. Test Results Compared with Elastic Beam Theory

	18 s.w.g. specimens	
	Specimen 1	Specimen 2
Deflection estimated to cause fracture	5.9×10^{-3} in	16.4×10^{-3} in
Measured deflection at fracture	0.5 in (av.)	1.117 in (av.)
Load estimated to cause fracture	0.0104 ton	0.0062 ton
Measured load at fracture ..	0.32 ton (av.)	0.33 ton (av.)
Estimated flexibility, K_h ..	0.57 in/ton	2.65 in/ton
Measured flexibility, K_h ..	1.30 in/ton	2.44 in/ton

The author has not evolved a complete plastic design method for thin tube plates, but suggests that the work involved could be done relatively easily provided that a systematic series of experiments could be made.

Fig. 9.—Static deflection test equipment with (below) specimens after failure.



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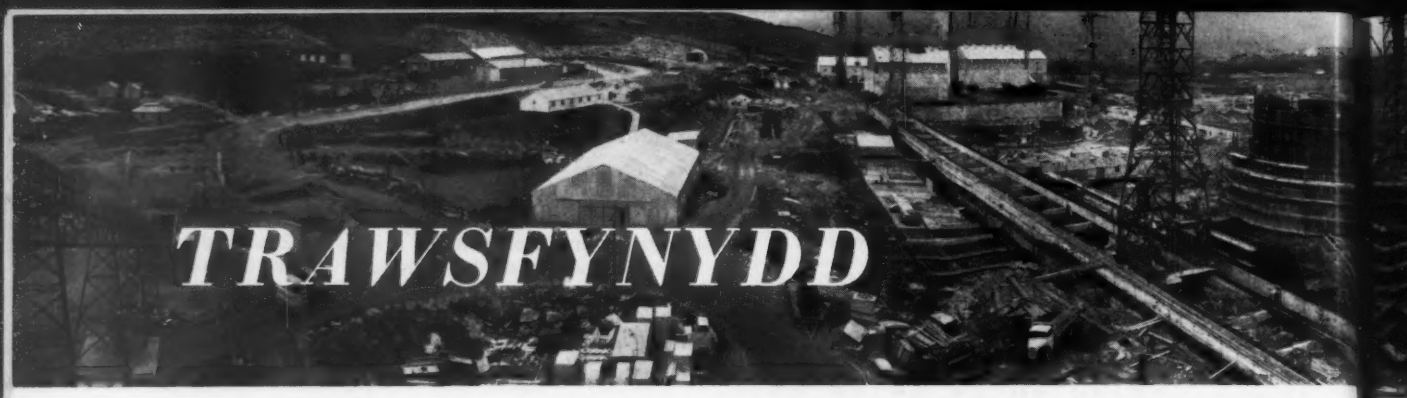


TRAWSFYNYDD

General Review of Scheme

NUCLEAR power stations are no longer "magic boxes." The sensation of yesterday becomes the commonplace of today and—so far from filling an entire journal with a minute description of every detail of a new and unfamiliar work—there is a tendency to regard each new station as "another gas-cooled reactor," and to dismiss it with the (already) time-worn cliché, formerly applied to the steam plant, ". . . of fairly conventional design. . . ." Nowadays, the focal points of interest lie where the design team have managed, either by bold strokes or sheer persistence in design refinement, to raise the efficiency or reduce the cost by a few per cent.; the emphasis is not upon the similarities, but on the differences.

Trawsfynydd, fifth of the "commercial" power stations, has many points of interest. Siting, cooling, civil work, coolant control, fuel handling, steam plant—all have presented fresh problems for solution by the design team of Atomic Power Constructions, Ltd.

The station has an installed capacity of 580 MW, and a guaranteed output of 500 MW. A brief description appeared in *Nuclear Engineering* for July-August-September (the composite issue), 1959, p. 289; full details are shown on the pull-out drawing in this issue.

Siting

Trawsfynydd is the first of the commercial stations not situated on the coast. It is, in fact, some 11-12 miles from Portmadoc, the nearest seaport, is about 600 ft above sea level and, being situated in the Snowdonia area (a national park), is subject to many additional restrictions from which previous stations have been free. The site is in Merionethshire, by Lake Trawsfynydd, an artificial reservoir originally constructed for the Maentwrog hydro-electric scheme, approximately three miles long and one mile wide.

Civil Engineering Work

No nuclear station is without its difficulties, and Trawsfynydd is no exception. As is usually the case, a great deal of work had to be carried out before proper access to the site was possible. The harbour at Portmadoc, once the principal shipping centre for Welsh slate, had deteriorated with the decline of the slate industry, and it was necessary to rebuild the quay and provide cranage for the heavy loads. Motorists who have toured in the rural districts of North Wales will need no reminding of the utter impracticability of transporting heavy loads along the narrow winding roads—often little better than lanes running between stone walls—and an extensive reconstruction programme of wider roads with more uniform gradients was necessary.

The rock site, while not presenting many of the difficulties of clay or shingle, nevertheless produced its own problems, being subject to a good deal of surface cracking; it was necessary to "clean up" for quite a depth before reaching sound rock. In this work, use was made of a comparatively new technique of air lancing; high-pressure air jets not only removed broken rock but dried out water in the crevices.

The restrictions imposed by the nature of the site not only dictated the architecture but extended even to the interior of the station. Every effort has been made in the station architecture to intrude as little as possible on the surrounding scenery; brushed concrete with the local aggregate makes the facings blend with the colour and striations of the rocky outcrops surrounding the site. Constant endeavour has always to be made to ensure that the site appearance is marred as little as possible even during construction; those who realize what any building site usually looks like during the early stages of construction will appreciate the task set.

No. 1 reactor area, from Goliath crane.

Not the least interesting detail of the station design is the cooling water supply. Cooling towers were inadmissible, and the condensers are cooled by utilizing the general mass of water in the body of the lake. What has not been previously emphasized, however, is the method by which dams, between small islands in the lake, provide a barrier almost completely across the lake and ensure that warm water discharged from the cooling circuits has to make a circuit of almost the entire lake perimeter before reaching the intake point.

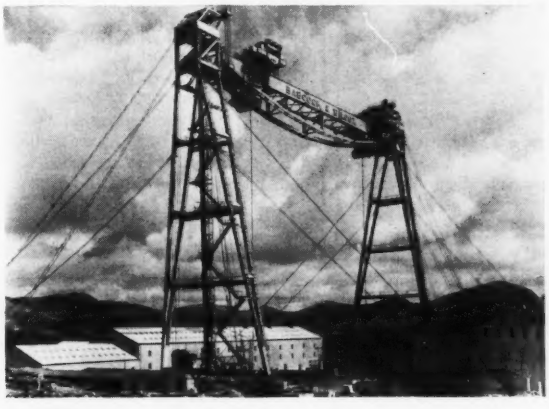
Reactor

Although the station shows an increase in overall efficiency and a reduction in vessel dimensions, the improvements are due to painstaking refinement of detail design rather than to the

General view of reactor and turbine house, from Goliath crane.

discovery and exploitation of any radically new principles. The gas pressure has been increased to 240 p.s.i. and the mass gas temperature to 399°C; more exact methods of calculation, made possible by extensive use of digital computers, have enabled the reduction of "uncertainty" margins, while still keeping within the temperature limits imposed by the Magnox canning. The design can temperature is 440°C.

A further advantage possessed by the Trawsfynydd design is the elimination of sleeves in the fuel channels; the design originally provided for their incorporation if necessary, but it has now been established that the graphite temperatures are sufficiently high to prevent unsafe accumulations of stored energy without their use. This modification not only simplifies



The 400-ton Goliath crane (Babcock and Wilcox) in course of erection.

construction, it has also the advantage of not presenting large quantities of irradiated graphite for disposal when refuelling. The fuel elements are of normal design, with Magnox canning having spiral fins and four longitudinal splitters reinforced with W-braces to prevent flutter.

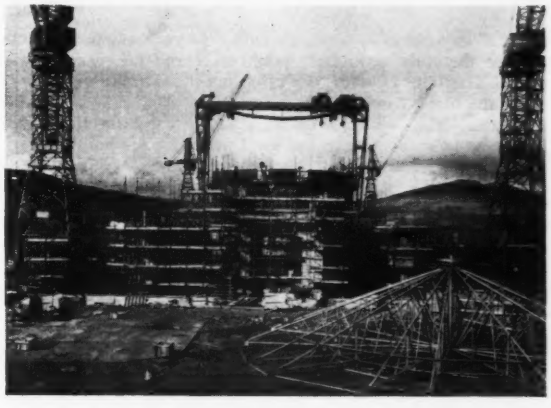
Reduction in size of the vessel to 61 ft internal diameter has been possible by the increase of pressure and temperature as well as an improvement in the space factor of the core (i.e., the percentage of the total volume occupied by the core) achieved by improved methods of core support.

As would be expected, the temperature control of the vessel supporting skirt is of considerable importance in minimizing stresses due to thermal expansion; it is interesting to note that exhaustive tests have been carried out on a one-third full-size model, under a wide range of working conditions.

Coolant Control

The coolant circuit shows a major departure from previous practice. For the first time, variable-speed drives for the blowers have been abandoned, in favour of direct-coupled induction motors.

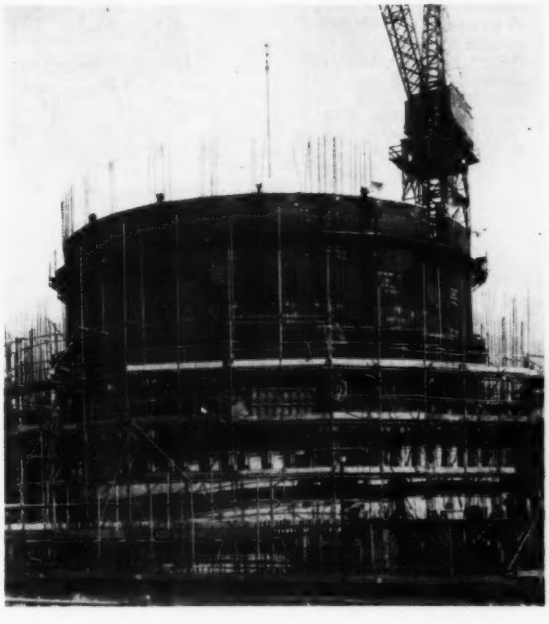
Temperature control, which would otherwise be achieved by varying blower speed, is here carried out by a combination of throttling of the inlet gas to the reactor, and recirculation of a portion via a by-pass valve, through the lower part of the heat exchanger. The throttle valve of the butterfly type, operates in opposition to the by-pass valve (i.e., one opening as the other closes), and the combined movement, operated by the main station control, gives practically constant gas inlet temperature. Variations in steam pressure or generator load first affect the gas valves, after which the outlet temperature causes variation in reactor output by control-rod movement.



No. 1 reactor area showing boiler plinths under construction.

A small amount of control for occasional use is provided by adjustable inlet vanes on the blowers. An auxiliary by-pass system is provided for start-up purposes. The throttle valves cannot completely close to allow on absolute shut-down, adequate decay heat removal by natural circulation.

A closer view of No. 1 reactor showing pressure vessel under construction.



The blowers are of interesting design. Of the single-stage axial type, they have spherical casings with their many well-known advantages: lower stresses; more even stress distribution; all openings and sections of true circular formation, thus facilitating welding manipulation and X-ray work.

The blowers, which are directly coupled to squirrel-cage motors at 2,950 rev/min, are fitted with flywheels to give approximately 20 min before shutdown, upon failure of supply.

Gas Ducting

The main gas ducting is 5 ft 6 in diameter. The main feature of interest lies, however, not in the ducting itself, but in the expansion joints, a new type designed by Richardsons Westgarth. A preliminary description appeared in *Nuclear Engineering* for October, 1960, p. 483, and it will be remembered that the two halves of the joint are connected together by two sets of high-tensile steel rods, arranged round the periphery so that each set forms the frustum of a cone. The joint behaves as a stiff connection in the axial direction, but angular movement is accommodated by pure elastic deformation of the rods. accurately predictable and not varying with time. The joint contains the usual bellows and floating sleeve. Since the forces acting upon the rods are accurately calculable, it is possible to obtain uniform stress distribution along their length by cold-swaging to the ideal profile.

The main advantage of this type of joint, of course, is that it is equally flexible in every direction, and there is no chance of troubles occurring due to seizure, as might possibly be the case with pin-jointed designs.

As will be observed from the pull-out drawing of the station, the outward appearance of the joint has changed since it was originally illustrated in these pages. The ring anchoring the outer ends of the two sets of rods has been reduced in width, thus moving the point of intersection of the two conical surfaces nearer to the bases of the cones, and making a more compact arrangement; the operation of the joint remains unchanged.

Fuel Handling

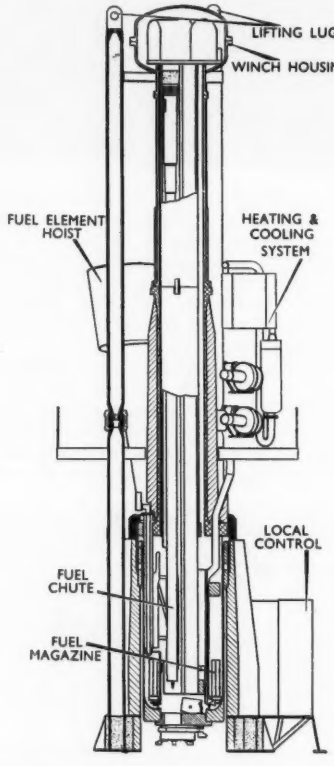
The philosophy of fuel handling differs from previous stations in that the machines are self-contained but not self-propelled. Also, although there are two machines, neither is required to work continuously, nor is one reserved purely as a spare. It is considered that the fuel cycle can be maintained with ease with "one-and-a-half" machines. In other words, one-quarter of the working time is available for maintenance, if required.

Each of the two machines is approximately 64 ft high, and weighs some 380 ton. Movement around the pile cap is by means of the charge-room crane, of the gantry type, which connnects to the machine by three links attached to motor-driven screw jacks which provide the short lift necessary to clear the charge floor. Service connections to the machines for power, compressed air, purge gas, etc., are made by means of a service trolley which carries cables and hose reels and enables the charge machine to be "plugged in" to service boxes loaded at strategic positions on the charge floor, eliminating long trailing pipes and cables from the machine.

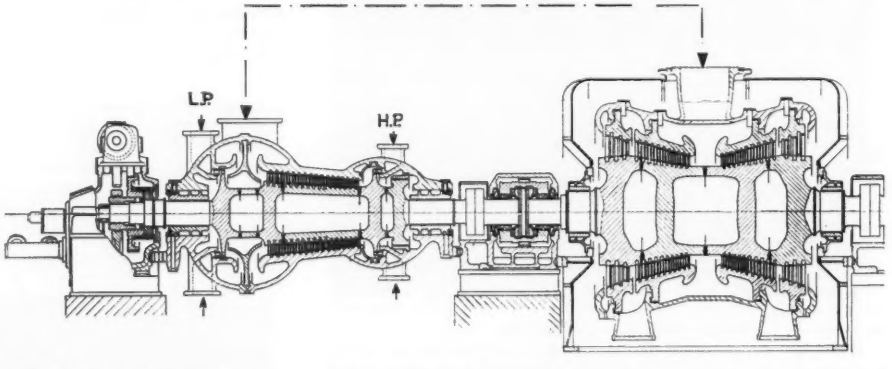
The central feature of the machine is a pressure vessel, with the usual arrangements for sealing to the selected stand-pipe, gas purging, etc. Inside this is a rotating chamber of trefoil arrangement, having three compartments, any one of which can be rotated over the standpipe. Two of these are for control rod servicing, one being empty, to receive the rod removed from the reactor, and one containing a new rod ready for insertion; both these chambers have their own hoists. The third compartment houses the fuel chute which is lowered by means of its own hoist; this is afterwards disengaged and withdrawn leaving the top of the chute free for the fuel grab to pass through. Other mechanisms, when the chute has been lowered, provide for rotation about the indexing hole, extension and retraction of the auxiliary chute on the end of the main chute, and positive lowering of the nose of the auxiliary chute right into the fuel channel so as to prevent blow-back troubles.

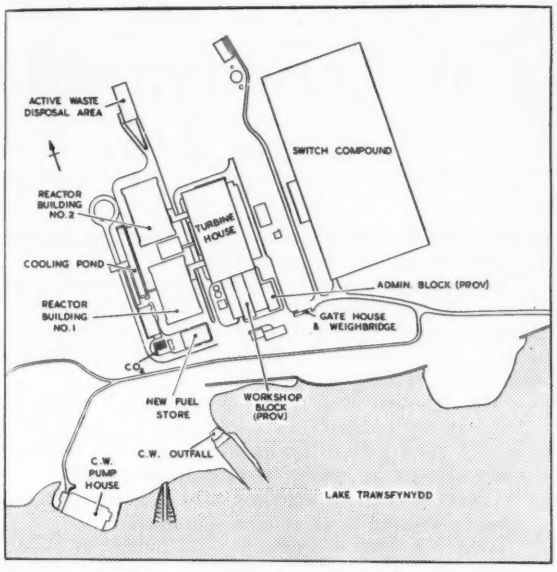
The actual fuel hoists, of which there are two, are enclosed in tubular housings offset from the centre line of the pressure vessel, and joining it at an angle. Between the fuel hoist housings and the central vessel, there is a magazine with an annular

Simplified section of the Fairey Engineering charge/discharge machine.



Section of the Richardsons Westgarth 145 MW turbine showing (l. to r.) H.P. cylinder with L.P. steam inlet, I.P. double-flow cylinder, and the twin triple-flow L.P. cylinders.





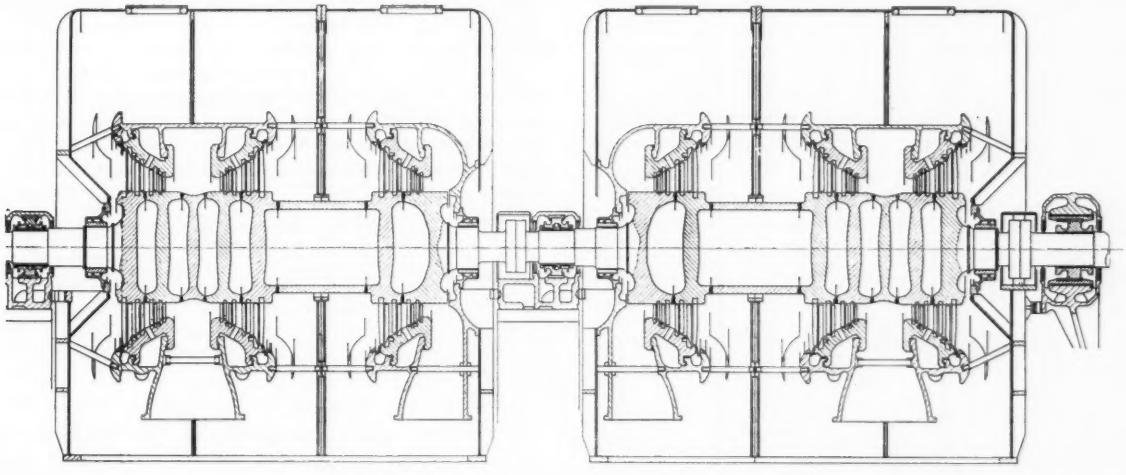
(Above) Heat exchanger shells in the International Combustion site fabrication shop. (Left) Site plan, indicating the end of the dam between intake and outfall.

turntable containing two staggered rings of tubular containers, each of which holds a fuel element. A transfer mechanism guides the fuel grab over the magazine or the mouth of the fuel chute, as required.

The magazine carries a total of 72 containers, arranged 36 in each ring. The same route is used for both fresh fuel and irradiated elements; the containers may be used for either, as required. Since there are nine elements in each channel, this means that—starting with nine empty compartments to receive the irradiated fuel from the first channel—seven channels can be serviced before the machine need be removed; a considerable advance on previous designs.

The machine is fitted with the necessary lead and water shielding, and with built-in cooling for the irradiated elements; this feature can also be applied to warming fresh fuel elements to the working temperature before loading into the reactor.

At the Heston factory of Fairey Engineering, Ltd., who are building the charge/discharge machines, an environmental test facility is under construction; it has heating and pressurizing facilities, and a standpipe extension so that detail developments can be tested under actual working conditions.



Steam Plant

Instead of glazed enclosures and local shielding, the boiler house has complete concrete shielding, with V-section ventilating louvres to prevent streaming. The heat exchanger design is based on heat transfer tests by International Combustion, Ltd., in a purpose-built test rig in which either air or CO_2 can be circulated at full working pressure and velocity. The shells are 116 ft high and 18 ft internal diameter; the thickness is $2\frac{1}{2}$ in. The finned tube sections are arranged in eight banks for (top to bottom) H.P. superheater; L.P. superheater; H.P. evaporator (two banks); H.P. economizer; L.P. evaporator (two banks); mixed (H.P./L.P.) economizer.

The turbine sets are outstanding in that less than a decade ago they would have been considered large sets for 1,500 p.s.i., 1,050°F steam conditions. Rated at 145 MW, the turbines are four-cylinder sets, with one H.P., one I.P., and two L.P. cylinders. The H.P. cylinder would, perhaps, be more correctly termed a mixed-pressure cylinder, since it receives steam at both pressures. The H.P. steam enters through a single velocity stage, followed by several stages of reaction blading, and mixes with the L.P. steam which enters at the other end through a single velocity stage. The steam then passes through the I.P. cylinder, a double-flow reaction unit with two exhausts and, finally, through two separate triple-flow L.P. cylinders each with its own separate condenser. Three-stage bled steam feed heating at 220°F is provided.

Moisture separators are located in the piping runs between the I.P. and L.P. sections; at this point the moisture content is about 11%. The low-pressure cylinders are provided with normal blade guards and drainage channels. At normal full load, the back pressure is about 1.2 p.s.i.a. (28.7 in vacuum).

Radiation Shielding at Trawsfynydd

by M. D. CLARK, M.Sc., A.Inst.P.
(Radiation Physics Group Leader, Atomic Power Constructions Ltd.)

IN the design of shielding for a nuclear power station, adequate attenuation must be provided of the radiation from a variety of gamma and neutron sources. For gamma sources such as irradiated fuel elements in the cooling pond and active control rods in the mortuaries, well-established methods of calculation are available. Uncertainties here are due to difficulties in describing source geometries and assessing self-shielding effects and in the estimation of build-up factors for cases of oblique incidence.

The calculation of neutron attenuation through bulk shields, is based on methods[1] developed by the Shielding Group at AERE, Harwell. They go a long way towards enabling the shielding physicist using only basic cross-section nuclear data, to deal with any combination of materials shielding any type and configuration of neutron source. Subject to the radiological safety requirements, emphasis throughout is placed on the achievement of minimum overall cost by good layout and the choice of economical combinations of shielding materials. This is especially important in the shielding of mixed neutron and gamma sources and is illustrated by considering the main biological shield and the fuelling machine shielding.

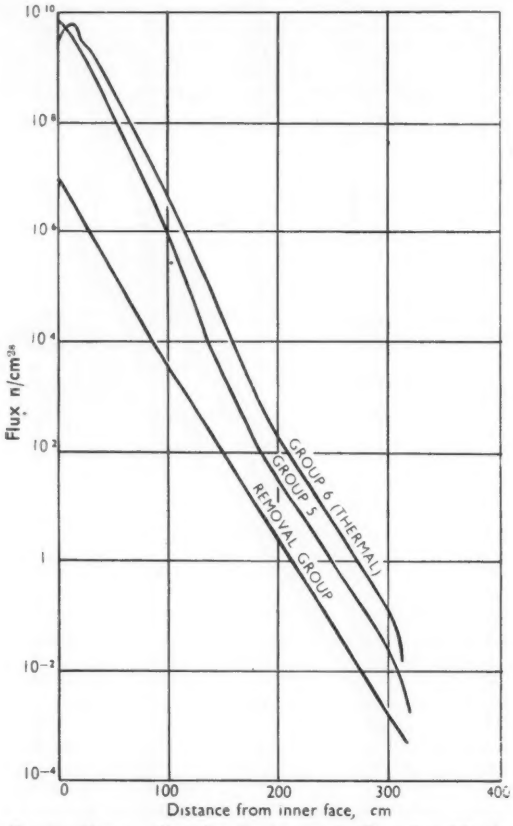


Fig. 1.—Neutron flux distribution in the Trawsfynydd side biological shield.

The radiological safety requirements specified by the Central Electricity Generating Board for Trawsfynydd are based on Recommendations of the International Commission on Radiological Protection[2] and they can be summarized as follow:—

The radiation level must not exceed 0.25 mrem per hour at any normally accessible point in the open or at any point within the buildings to which personnel not employed in reactor operation have access.

Transient dose rates greater than 50 mrem per hour must be avoided in all routine operations.

Annual integrated exposures from routine or planned operations, including inspection and maintenance, must be less than 2.5 rems from all sources of radiation.

Throughout the design the recommended methods and data[1] of AERE-R3216 are employed and extensive use is made of digital computer programmes written either within A.P.C.[3] or by the Shielding Group at Harwell[1, 4, 5].

The method of calculating neutron flux distributions through the main biological shield is fully described elsewhere[1] and only a brief account is given here. Fluxes are divided into seven energy groups as indicated in Table 1: a "removal" group, five intermediate groups and a thermal group. The removal group consists of high energy neutrons in the fission range of 1-14 MeV which can penetrate deep into the shield due to the comparatively low nuclear cross-sections at these energies: flux levels across the shield are obtained from the fission source distribution in the core using energy-dependent removal cross-sections. Neutrons of intermediate energy arise either directly from the reactor core or by slowing down from the removal group and flux distributions for these and for the thermal neutrons are obtained using conventional multi-group diffusion theory.

TABLE 1—Neutron Energy Groups

Group No.	Energy range
0	Removal
1	2.0 – 0.3 MeV
2	0.3 – 0.03 MeV
3	0.03 – 0.01 MeV
4	0.01 MeV – 100 eV
5	100 – 0.0711 eV
6	Thermal

Of the two sources contributing to the intermediate and thermal energy groups, that due to the removal group predominates at penetrations of more than 10 slowing-down lengths in most materials. Thus, in a homogeneous medium, the various energy groups achieve equilibrium, and decay thereafter with the same attenuation length. This process is illustrated in Fig. 1, which shows the neutron flux distribution in the Trawsfynydd side biological shield for the removal group and groups 5 and 6.

Main Biological Shielding

Aggregate available locally at Trawsfynydd is not of particularly high density and its use in preference to heavier material from another area necessitates the provi-

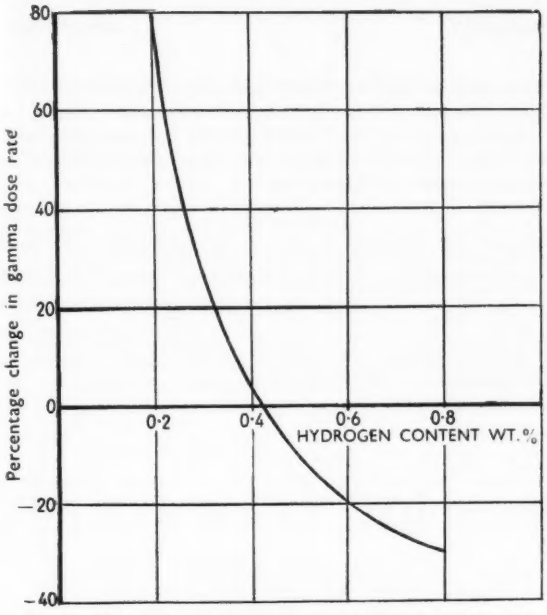


Fig. 2.—Variation with hydrogen content of gamma dose rate at surface of side biological shield.

sion of thicker shielding and longer ducts and standpipes. Transport savings, however, amply justify this choice and the concrete mix in use has the chemical composition given in Table 2 with a mean dried out density of 142 lb/ft³.

TABLE 2

Chemical Composition of Dried-out Shielding Concrete, wt%

Silicon	32.34
Aluminium	4.79
Iron	1.83
Calcium	5.86
Magnesium	2.18
Sulphur	0.14
Alkali (Na+K)	0.49
Oxygen	51.44
Hydrogen	0.42
Carbon	0.51

Since the hydrogen content is low its effect on the thickness of the side biological shield and the dose rate outside it has been calculated and the results are presented in Figs. 2 and 3. The dose rates are almost entirely due to capture gammas from the pressure vessel and concrete, the contribution from neutrons being less than 1%, and it can be seen that even if an increase of 50% in hydrogen content were achieved the shield thickness could only be reduced by 2 in, i.e., 1½% of the total thickness and of negligible significance.

The biological shield surrounding the reactor is a cylindrical concrete structure with walls 10 ft 6 in thick up to the 43 ft 9 in level and 10 ft thick above this. All apertures are protected, either by secondary shield walls as in the case of the main gas ducts and the BSD ducts, or by muffs occupying the annuli surrounding such items as standpipes and Wigner probes.

The reactor top shield is 11 ft 6 in thick and there is a 2-ft space between it and the pile cap occupied by cable ducts and standpipe cooling channels. A steel-shot concrete grid in this area limits radiation to acceptable levels during the discharge of spent fuel.

Shielding of the gas circuits is effected by the complete and separate enclosure of each main duct and its associated boiler. The concrete walls surrounding a boiler vary in thickness from 1 ft to 1 ft 6 in, but all equipment necessary for the routine operation of the plant, including steam drums and circulating pumps, is located externally in areas of unlimited access. This arrangement also facilitates work on a shut-down boiler when the maintenance staff will not be exposed to abnormal radiation.

Fuelling Machine Shielding

The multi-purpose Trawsfynydd fuelling machine is designed for the preparation of standpipes for charge-discharge operations and for changing control rods, in addition to its main function of handling fuel elements, all operations being carried out under full power conditions. Shielding of this complex machine, illustrated in Fig. 4, must provide adequate attenuation of the radiation from three main sources:

Gammas from irradiated components such as control rods and the fuel chute.

Gammas and delayed neutrons from irradiated fuel elements.

Gammas and neutrons streaming from the core and scattering within the machine whilst it is over an open standpipe.

The base of the machine is shielded by lead and concrete and supported on a rubber pad. To reduce gamma streaming through this pad a floating steel collar fitted at the inner diameter of the shield rests on the pile cap when the machine is lowered on to a standpipe.

At the fuel magazine level the main shielding is lead and borated water while immediately above this the pressure vessel diameter decreases and the borated water is replaced by compressed wood. Choice of the lead-borated water arrangement is the result of an optimization study for

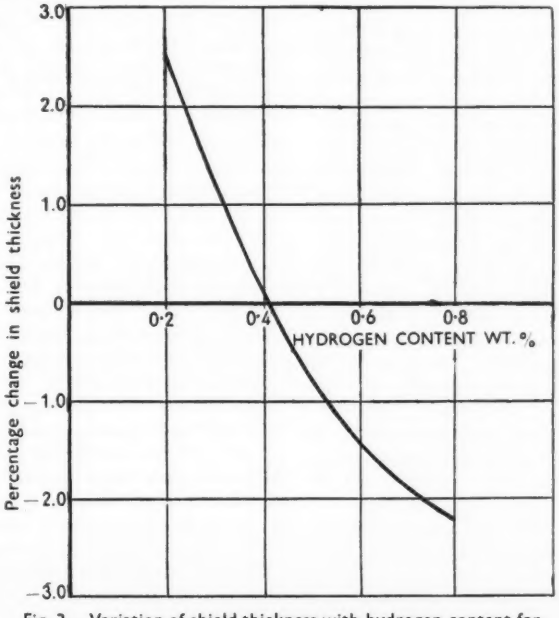


Fig. 3.—Variation of shield thickness with hydrogen content for constant gamma dose rate.

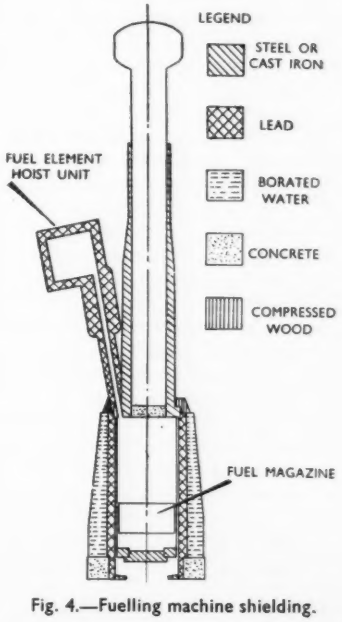


Fig. 4.—Fuelling machine shielding.

a minimum weight shield. Various combinations of lead, steel, concrete, compressed wood and water have been considered and the lead-borated water system shows a weight advantage for a given dose rate of more than 10% over its nearest rival.

Significant penetration of neutrons into the upper section of the machine, which contains the fuel chute and control rods, is prevented by filling the interstices between the stowage tubes and the pressure vessel with concrete 1 ft thick, and by the positioning of a concrete plug of similar thickness when the fuel chute is lowered. Shielding is, therefore, only required against gamma radiation, and this is provided by cast iron, except around the detachable fuel element- and television camera-hoist units where lead is used because of the limited space available.

Optimization of the main shielding and the lightness of the upper structure keep the overall weight of the fuelling machine below 400 tons.

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Fault Conditions Analysis for Trawsfynydd

by G. M. BRACEWELL, A.M.I.E.E.

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THE analysis and assessment of the sequence of events under all credible fault conditions of the reactors and associated plant are important aspects of the overall design and need to be taken into account at the earliest possible stage so that an appropriate balance is achieved *ab initio* between economy, performance and safety.

Objects of the Fault Study

Further ways in which the fault analysis influences the design concepts of important plant items are as follows:—

1. To provide a basis for the emergency procedures to be adopted during fault conditions, taking due account of environmental changes which may attend severe faults;
2. To determine what emergency equipment and auxiliaries are required, in terms of effectiveness and availability;
3. To establish beyond any reasonable doubt that the automatic shut-down system is capable of shutting down the reactor, under all credible fault conditions, in such a manner that time is available to ensure the effectiveness of emergency procedures;
4. To provide a description of reactor behaviour in fault conditions from which the overall reliability (defined here qualitatively as the ability to continue functioning under any credible state of self-failure) required of the automatic shut-down system, alarms, protective apparatus and emergency procedures can be assessed.

In less general terms, the initial transient changes of neutron fluxes, temperatures and pressures in fault conditions determine the relative importance of individual trip parameters providing protection against particular faults, together with the performance required of detectors, electrical circuits and actuators. This provides a basis for adjusting the design so that the probability of short time-scale transients causing a major reactor hazard is reduced to an acceptably small value. Similiarly, the progress of the longer time-scale transients following a fault is of vital concern in the design of auxiliary forced cooling facilities and carbon dioxide make-up supplies, in the formulation of realistic emergency procedures, and also in determining the extent of environmental hazards, if any, likely to be encountered by the operators in correcting the fault.

Thus, considering the initial action of shutting down the reactor following a given fault, the general problem may be resolved into determining:—

1. The total time available before particular reactor parameters (e.g., can temperature) reach particular limiting values which must not be exceeded;
2. The time elapsing before each trip parameter reaches a pre-set level and initiates shut-down action;

hence determining the time margin available between the trip set point and the so-called "damage point" of the reactor; and indirectly the standard of performance and sensitivity required of the automatic shut-down system.

Considering the more severe fault conditions that can be postulated, it may be that the initial action of shutting down the reactor merely arrests the approach to a dangerous condition, in which case further action by the operator may be required—e.g., closure of valves, restoration of some measure of forced cooling—and it is necessary, therefore, to determine the time available to initiate these procedures.

The foregoing comments represent the general aims of the fault analysis in furthering the design of the Trawsfynydd reactors. Detailed methods of calculation will not be discussed here, first, because of space limitations and, second, because many of the methods are at this time under development in order to obtain more detailed representations of reactor dynamic behaviour necessary[1] in assessing the overall safety of Stage I reactors. It is proposed to confine the remainder of this article to some brief comments on the more important problems for which new methods of computation are now being developed.

Faults Causing Divergence

The question of the time and space separability of the kinetic equations is being carefully considered in relation to the need (or otherwise) for neutron sources during initial operations, when the control rods are withdrawn in groups at low power level. As part of a general review of the applicability of existing methods of computing start-up transients, consideration is also being given to the effect of distortions of the main flux and temperature distributions, when divergence occurs due to movements of partially inserted rod groups. The overall objective of this work is to obtain a better assessment of potential problems during commissioning and subsequent start-up, i.e.:—

1. To determine the sequence in which the trips operate;
2. To establish beyond reasonable doubt that the reactor will be safely shut down if it diverges from any shut-down condition even if the "first" trip has failed;

3. To show whether or not a pre-irradiated neutron source is required during the early life of the reactor;
4. To ensure that the trip settings are correctly chosen.

Parallel investigations are being made of the stability, response, and accuracy of trip system equipment at the count rates corresponding to the various trip levels, e.g., in relating the period trip settings with computed transients.

The effect of partial instability of sector and station automatic control systems under fault conditions is also under intensive investigation.

Channel Fires

Various methods have been employed in the analyses of this fault; existing methods, however, do not account for the effects of the gas spaces between blocks of graphite and the flow in adjacent channels, and it is considered that these limitations tend to give a pessimistic estimate of temperature changes in adjacent channels, and thus of the risk that the fire will spread laterally. A method is under development which attempts to account for these effects with the object of determining:—

1. The amount of Wigner energy which can be stored in the moderator before there is any risk of the fire spreading to adjacent channels;
2. The time margin available in which to detect the fire and shut down the reactor, in order to prevent lateral spread of the fire.

Methods of combating single or multiple channel fires are also under investigation.

Failure of Gas Circulators

The general object of these computations is to determine the variation of uranium, can, coolant and moderator temperatures (also gas pressure changes) with time, following partial or total failure of gas circulators, and to estimate the effect on trip operation of transients occurring (e.g., during start-up) when the flux and temperature distributions may be distorted owing to partially inserted control rod groups so that, although temperatures in the active region of the core may be about normal, those in the upper region may be depressed. Bearing in mind the probable location of trip detectors in the top part of the core, it will be necessary therefore to investigate this effect to ensure that the temperature trip settings will be satisfactory should partial or total loss of flow occur under these conditions.

Clearly, both axial and radial temperature distributions ideally require to be computed for the more rapid transients. The main purpose of this calculation is to assess the performance of the reactor shut-down system, when initiated by trips having temperature-dependent sensing elements, so that, notwithstanding the provision of fast response trips associated with failure of electrical supplies to the gas circulators, the effectiveness of subsequent slower trips can be assessed. (A further object is to determine the maximum trip time lag that can be tolerated.)

In the case of total failure of forced gas circulation, the flow changes appropriate to loss of electrical supply to all circulators is generated as a time-dependent function, terminating in natural circulation at the appropriate gas pressure. In this fault, the first trip to operate will be that connected to the 11 kV Unit Supply Boards which supply the circulator motors. The second and third trips to operate will probably be those associated with excess fuel element and bulk gas outlet temperatures respectively.

Since the transients are computed for a range of prompt negative temperature coefficients to take account of possible changes in the latter due to prolonged irradiation, the computations for the more rapid transients associated with smaller values of the prompt coefficient include a representation of the estimated shut-down action, i.e., the estimated time-dependent reactivity-worth function of the control rods is inserted at time t where t is the sum of, for example:—

Time to reach detector trip level + detector and amplifier lag (referred to step response) + safety circuit and contactor lag + any time lag whilst rods drop from banked position into core region.

A further factor in the case of gas temperature trips is the dependency of detector time lag on gas velocity.

Fig. 1 illustrates the effect on the time available for operation of temperature-dependent trips, following total or partial loss of forced circulation (with no gross pressure loss), of changes in the value of the prompt negative uranium temperature coefficient over the likely range of values which might be assumed.

Gross Pressure Loss

The probability of a fault occurring of such magnitude as to cause a substantial breach of the pressure circuit is exceedingly remote; nevertheless, it is necessary to consider the sequence of events and actions to be taken in dealing with this fault which, by its severity, is generally regarded as the limiting credible accident to the reactors.

The analysis and assessment of the consequences of a rapid pressure loss involve many considerations, and space limitations necessitate some condensation of the arguments. Briefly, two main requirements are placed on the design of the reactor and its auxiliary equipment:—

1. The automatic shut-down system must be capable of shutting down the reactor *with certainty* in a sufficiently short time to ensure that the initial temperature transients do not cause components (e.g., fuel cans) to exceed a limiting safe temperature in the period before some measure of heat removal from the reactor core is restored; the coolant flow provided must then be sufficient to keep these temperatures at safe levels thereafter. In this analysis the further

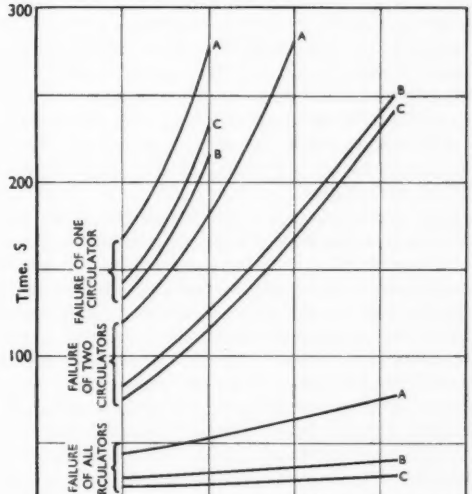

Fig. 1.—Time sequence of temperature trips following total or partial loss of forced circulation. (Assumes 11 kV trips have failed to operate and that sector servo system is inoperative.)
(A) Time for hottest can to reach 580°C with no trip action.
(B) Time to reach fuel temperature trip level (+50°C).
(C) Time to reach outlet duct gas temperature trip level (+50°C).

possibility that the coolant may be a mixture of air and CO_2, which reacts exothermically with graphite, must be taken into account;

2. The design must ensure that the facilities for preventing any spread of radioactive contamination or toxic gases are such that no general hazard can be caused.

Consideration of the transient changes in coolant pressure and flow immediately after a breach of the coolant circuit suggests that a so-called "limiting" case exists, namely, that a breach can be envisaged to occur at a particular position which, when associated with a particular size of breach, may cause the gas flow through the core to stagnate very rapidly because of the disappearance of any pressure differential across the core. A semi-qualitative study of the flow change indicates that the duration of stagnation is likely to be some 30 or 40 seconds after the breach.

Consequently, it is instructive to consider the changes in fuel can temperatures during this period assuming instantaneous and total loss of forced convective cooling. Clearly, heat will be transferred from the fuel to the moderator mainly by radiation and local convection, so that the combined can/graphite emissivity will be important. Equally, the total heat produced in the fuel during stagnation will depend, first, on the promptness of shutting down the reactor and, second, on the magnitude of fission product heating, so that trip time delay and time-dependent fission product decay will also be important.

Fig. 2 shows the initial rise of temperature with time of the hottest cans in the reactor following a rapid loss of circuit pressure according to the foregoing assumptions. The fission product decay heating assumed is based on that reported by Stehn and Clancy[2] multiplied overall by appropriate factors, marked on Fig. 2, which also give various shut-down time delays and an appropriate range of emission coefficients. The computed curves are slightly pessimistic as it was assumed that only 6% of the fission product heat appears in the moderator; the effect of assuming 15% in the moderator is to reduce the rate of temperature rise, e.g., for the typical curve marked (1.4)(3)(3) Fig. 2, the can temperature would be 5°C lower at 10 seconds, 7°C at 40 seconds, and 10°C at 100 seconds. Can temperatures reached in times greater than 35 seconds will be somewhat hypothetical on the grounds stated in preceding paragraphs.

Clearly, the physical and mathematical model will be "pessimistic" as regards the assumption made about the nature of the initial flow change; nevertheless, a few tentative conclusions can usefully be made, based on an arbitrary assumption that at the end of the postulated stagnation period the temperature of the hottest cans must not exceed the lower limit of 580°C, which is quoted for illustrative purposes in Ref. 3. First, it would appear that a 5-sec trip delay cannot be tolerated, no matter what assumptions are made regarding the magnitude of radiative heat transfer, within the range of magnitudes of fission product heating investigated. Second, if the maximum credible multiple of fission product heat decay curve is assumed to be 1.7, then a trip delay of 4 sec will only be tolerable provided the combined can/graphite effective emission constant per unit channel has a value approaching 6×10^{-11} kW/°C⁴(A), which is not considered a practical possibility, whilst lower values of the decay heat would put less stringent requirements on the can/graphite emissivity. Third, a time delay of 3 sec would enable the maximum credible pessimistic assumption to be made about the magnitude of decay heat

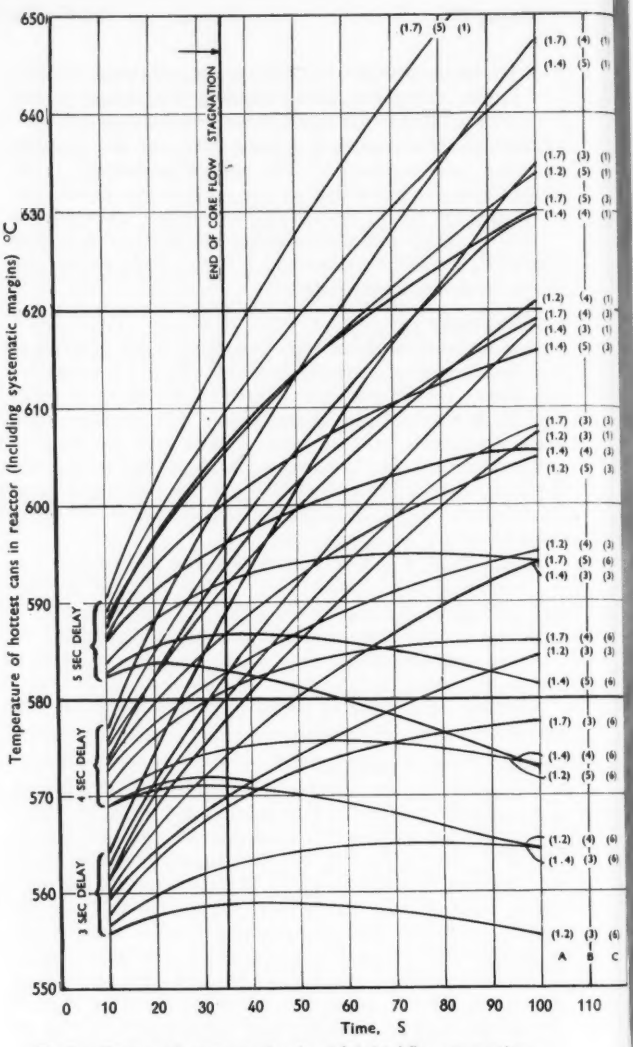

Fig. 2.—Fractured pressure circuit with initial flow stagnation. (A) Multiple of fission product heat curve. (B) Delay in shutting down reactor (seconds). (C) Heat radiation coefficient $\times 10^{11}$ kW/°C⁴(A) per unit channel.

provided the combined emission constant is not appreciably below 3×10^{-11} kW/°C⁴(A).

It would appear from this simple analysis that, using the criterion of 580°C as the limiting can temperature, and values of can/graphite emissivity likely to be encountered in practice, sufficient reactivity to shut down the reactor must be inserted within 3 sec of the pressure circuit burst if pessimistic assumptions as to the magnitude of the fission product decay heat are used. Clearly the use of a less severe, but not unreasonable, criterion of 600°C[3] as the limiting temperature makes the demand on the shut-down system less difficult. Whichever criterion is chosen, there appears to be a need for some form of quick-acting "secondary" shut-down system for dealing with a severe breach of the pressure circuit, if such an event is assumed to be credible.

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2. Stehn, J. R. and Clancy, E. F. "Fission-Product radioactivity and Heat Generation." Knolls Atomic Power Laboratory. G.E., Geneva Paper. P/1071/USA. September, 1958.
3. Gott, H. H. *et al.*, "A General Survey of the influence of heat transfer on the performance of gas-cooled power reactors." Paper 1. Symposium on the use of secondary surfaces for heat transfer with clean gases. I.Mech.E. Nov. 9, 1960.

Design of the Trawsfynydd Graphite Cores

The aim of the Trawsfynydd core design is to maintain the integrity of the channels under all working conditions and following any credible faults, such that control holes remain operational, fuel elements can be put in or taken out, and the coolant leakage from the fuel element channels is kept to a minimum. Channel distortions, as such, are not considered important, provided the conditions stipulated are met.

by P. C. WARNER, M.A., A.M.I.Mech.E.
(*Design Department, Atomic Power Constructions, Ltd.*)

EACH Trawsfynydd core has 3,740 vertical fuel element channels $3\frac{1}{2}$ in diameter, and another 465 vertical holes varying between 3 in and $3\frac{1}{4}$ in diameter which can be used for control rods, flux scanning, graphite monitoring, flattening bars and so on. The graphite is supplied as blocks up to 32 in long and the block cross-section is chosen to correspond to one fuel lattice pitch, so that a fuel element channel consists of a column of blocks stacked on end with their extrusion axes vertical; the channel itself is formed along the axis of the blocks, and not on their flanks, so that joint leakage is confined to the end faces. Control holes, for which leakage is not so important, are formed by fillets at the corners of blocks (Fig. 1).

The UTS of the graphite used in Trawsfynydd is of the order of 1,000 lb/in^2 and its UCS of the order of 4,000 lb/in^2. Scatter of these figures in production specimens may be substantial, owing to internal flaws or other causes, and so design stress levels are kept low (no economic flaw detection procedure exists). Irradiation tends to increase mechanical strength but corrosion reduces it, and the net effect is adverse.

It may be mentioned here that there are over 100,000 primary graphite components in each core, so that design stresses ought to be selected statistically, in the sense of keeping the probabilities of failure of given percentages of components below certain values. Not enough is known about the spread of ultimate strength to apply this principle to Trawsfynydd except in a general qualitative sense, but it will doubtless be the basis of design in the future and of new inspection techniques. There is also a wide scatter of permeabilities in production specimens of reactor graphite, possibly due to the random incidence of internal cracks.

Graphite's chemical reaction rate with carbon dioxide, which in the ordinary way is quite slow at reactor temperatures, is accelerated in the nuclear flux by several orders of magnitude. There are two engineering consequences. The first is that properties such as permeability and mechanical strength deteriorate with time so that the design values of these properties must be reduced to allow for it. The second is that deposition of carbon may be expected from the back reaction, and current experimental evidence is not adequate to predict with any confidence where it may occur.

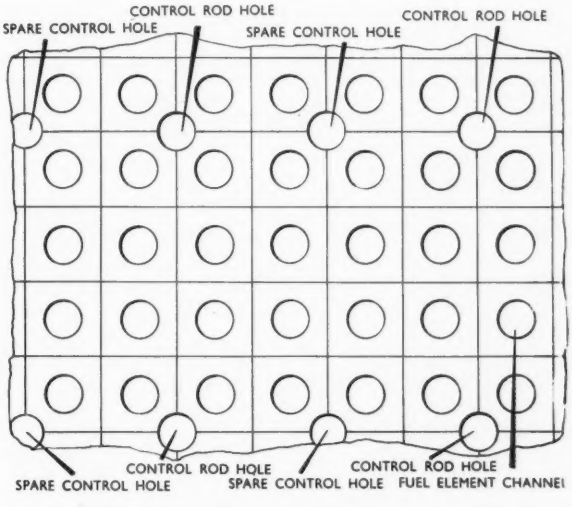


Fig. 1.—Control rod holes are formed by fillets at the corners of blocks.

Irradiation Strain

Depending on the temperature of irradiation, the total dose experienced, and the direction of extrusion, the strain may be growth or shrinkage, and will vary in magnitude. Although there exists a large amount of experimental data which can be used as a basis for design, obviously such information can never be exhaustive, and so the philosophy adopted from the start on Trawsfynydd has been to produce a design that would be insensitive to variations from the estimated Wigner strain. This at once excludes any principle which depends on predicting and subtracting strains with a view to achieving a small net effect, as for instance strains in two different directions relative to the extrusion axis.

Lateral Strain of the Blocks

Every block in a vertical column grows or shrinks laterally at every level, according to its mean temperature and dose. These vary; the temperature increases from inlet to outlet, and the mean flux has the usual cosine and flattened Bessel shapes. Lateral strain therefore varies up the blocks, and from column to column.

In the active core where the lateral strains are large, layers of blocks, 32 in high, alternate with layers of tiles, 4 in high, the tiles alone forming the structural skeleton at each horizontal level, leaving the vertical faces of blocks separated by gaps. In this way, each block can grow or shrink laterally without interfering with its neighbour; block machining is simpler, special structural features, or increased raw material sizes, are confined to the tiles, and the latest experimental data can be incorporated into their design, without interfering with block detailing and manufacture. Suitable end features keep the blocks concentric with the tiles, and stop them rotating, while in elevation the two ends of each block are effectively pin-jointed to the tile above and below and located by them.

There are two ways of dealing with the lateral strain of the tiles themselves. The first is to allow them to butt against one another and to clamp each layer solidly by inward forces at the periphery. The objection to this is that strains accumulate along a diameter, and location is decided by the total. Large growths in any layer push out the outside channels and cause kinks in them. Worse still, at layers where shrinkage predominates, the tiles near the centre of the core where it is highest go slack and lose their location. A system that integrates strains can

therefore be used only when these are predictable and small; this is not so with the Trawsfynydd cores.

Instead, the structural skelton is designed to keep each fuel column nearly vertical by making location independent of lateral strain. Here again there are a number of possible methods. Columns could be put in compression, but this is soon shown to be impractical owing either to the size of the necessary dead weight or to the poisoning effects and relaxation of any tie members. Methods in which tiles with sloping faces or other wedges move vertically suffer from thermal ratcheting. A framework of some material free from Wigner effects such as zirconium, lying within the tile layers, is costly and poisonous. The actual method used on Trawsfynydd is to interlock the tiles by keys with vertical faces, whose axes lie along the lines joining the centres (see Fig. 2). Such a layer principle has been used elsewhere, and has the property that when tiles change their dimensions, the foci (tile centres) remain fixed in space, because key and keyway undergo the same dimensional change without affecting clearance and move relatively only in the direction of their common axis. With a square fuel lattice, a single tile has eight keys to link it with its neighbours.

A point of interest is that the focus need not be at the tile centre. For example, tiles could be grouped in fours, and the common corner used as focus; then by staggering the groups, there would be only one key per tile as well as a collar to hold the four together solidly. However, relative vertical movements cause problems, and in the present design, tiles act singly. Face keys are made integral to keep down the number of components to be machined and handled, but the cruciform keys at the corners have to be separate. Fig. 3 shows a model used to investigate the properties of a tile layer.

The top and bottom blocks in the columns are reflector blocks and form solid slabs right across the core. Thus, when the temperature rises the columns are splayed outwards and, in so far as the mean temperature at each level is uniform, the keys do not slide. The tile layers are very stiff against all displacements except expansion or contraction as a complete circle, so must be held at the periphery. They are therefore spigoted to the side reflector, consisting of a double thickness of butting blocks forming a cylindrical wall of fixed diameter, and clamped by temperature compensated garters. It is the tile layers themselves that maintain the circularity. The whole core is on ball bearings.

The design just described therefore allows free lateral strain of blocks and tiles, reflector blocks being relieved where the dose is significant. Thermal expansion does not upset the stability or make the keys work.

Symmetrical Strain Variations in Blocks

The flux variation within a block has a symmetrical component whose contours are highest at the bore and fall off radially; the corresponding Wigner strains cause strains analogous to the thermal stresses in a thick cylinder with radial heat flow, and they are calculated in the same way. The picture is modified if there are also temperature contours, but the gradient is small and porous flow tends to reduce it still further. Graphite creeps under stress when irradiated, which tends to relieve such stresses.

Flux Gradients—Joint Face Leakage

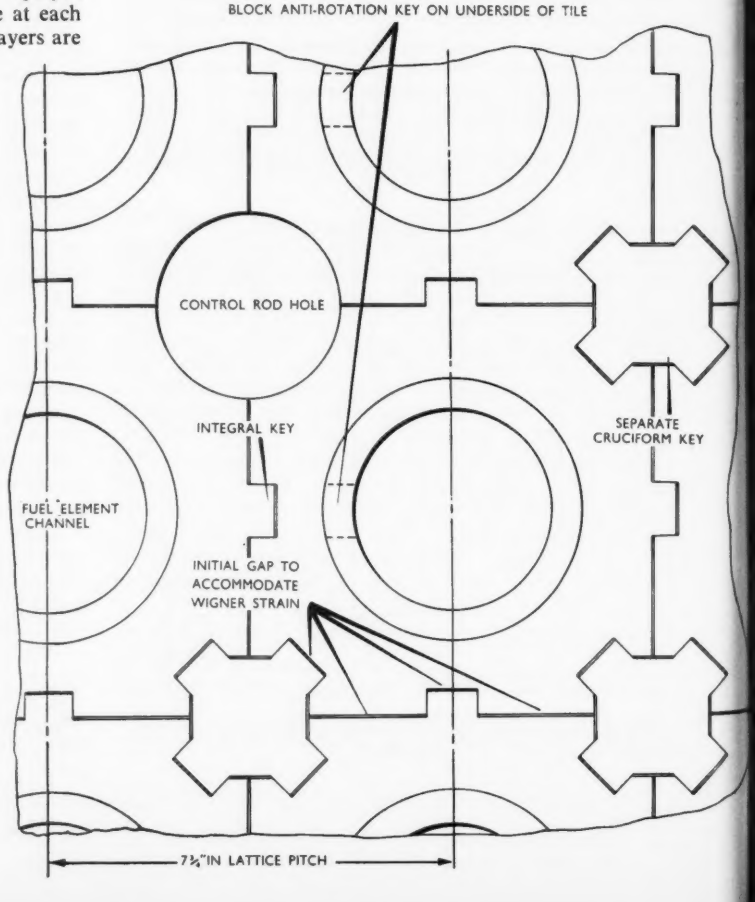
There are flux gradients across blocks near an absorber hole, and across all those in the unflattened zone. This causes uneven axial strains, so that originally parallel end faces become angled, and as the vertical axis is held by the tiles above and below, a block will now be resting on its edge and there will be a wedge-shaped opening between it and the tiles. This is one reason why laying blocks on their side, or calling for large extrusions embracing more than one fuel pitch, would not be a good way of designing a core; it would magnify such gaps.

To reduce both the offset of the load and the gap, the ends of blocks are faced circular. Around absorber holes the instability forces cancel out and the columns lean inwards, via their tiles, on one another. In the unflattened zone there is a small outward force transmitted through the tile layers to the garters.

Leakage is a more serious problem. In going from the space under the core, at inlet pressure, to the space above it, at outlet pressure, the gas can follow alternative paths; the objective is to constrict it to the fuel channels. Outside the core its way is cut off by the main gas seal; the control holes are blind at their lower ends, and the bottom reflector slab forms a sealing diaphragm right across, so that virtually all the gas must enter the fuel channels. Once past the bottom reflector, however, there is a leak path out into the gaps surrounding the core blocks and thence to the nearest control hole, which is vented to the top because that is the side from which control rods and so on enter.

In the core as new, leakage out of fuel channels is mainly porous, but when block end faces separate it could rise considerably (the separation is independent of the number of faces), and special seals have to be provided at the joints consisting of Magnox rings in annular grooves

Fig. 2.—Detail of interlocking system for moderator blocks.



The World's Reactors

No. 31 TRAWSFYNYDD

OWNER: Central Electricity Generating Board.

PURPOSE: Power production.

LOCATION: Trawsfynydd, Merionethshire, North Wales.

DESIGNER AND CONTRACTOR: Atomic Power Constructions, Ltd.

TYPE: Thermal, heterogeneous, CO_2 gas cooled, graphite moderated.

PROGRAMME:
Entry to site: July 1959.
First reactor and associated plant on load: October 1, 1963.
Second reactor and associated plant on load: July 1, 1964.

CAPACITY:
Net electrical output: 500 MW.
Number of reactors: Two.
Heat rate per reactor: 870 MW.

FUEL:
Material: natural uranium metal rods.
Dimensions: 1⅛ in diameter.
Elements in vertical channels—
Number per channel: 9.
Number of fuel element channels: 3,740.
Diameter of fuel element channels 3¾ in.
Density of uranium: 18.78 g/cm³.
Total weight of uranium: approx. 280 tons.
Mean fuel rating: 3.11 MW/t.

CANNING:
Material: magnesium alloy.
Extended surface: longitudinal spiral, 4 straight splitters.
Method of element support: stacked.
Maximum surface temperature: approx. 440°C.

MODERATOR:
Material—active core: graphite " A " type;
—reflector: graphite " B " type.
Active core—height: 24 ft;
—diameter: 44.5 ft.
Dimensions over reflector: height: 27 ft 3 in;
plan: 48 ft over 24 flats.
Total weight per reactor: approx. 1,900 tons.
Method of support: ball bearings on plates on grid strutted from internal skirt.
Restraints: circumferential.

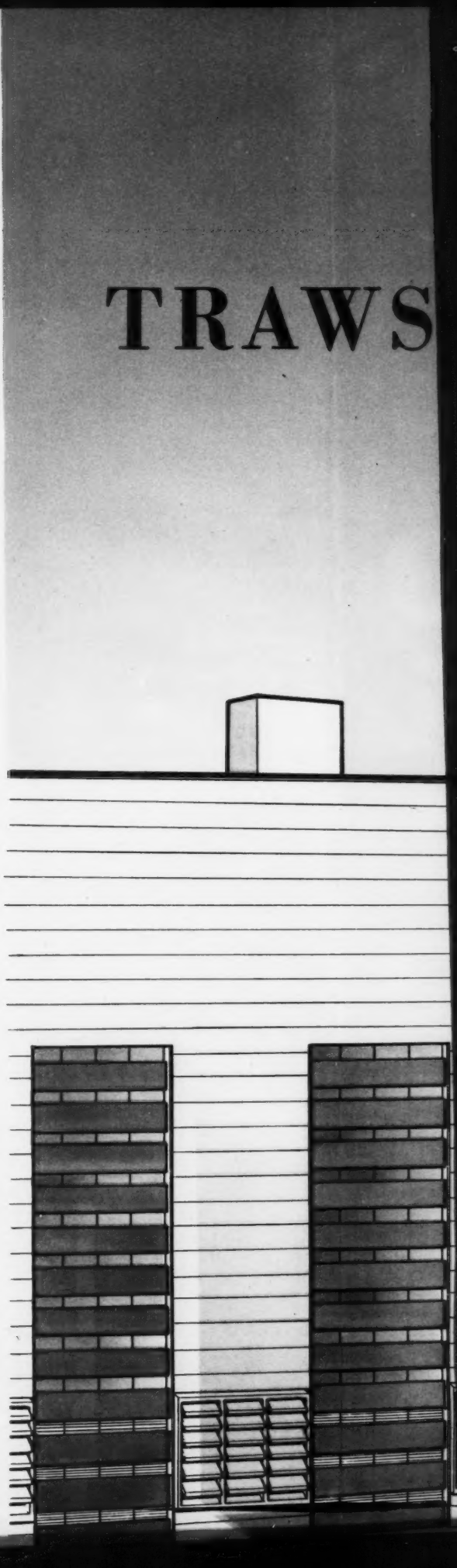
CORE DESIGN:
Average temperature of graphite: ~310°C
Lattice—arrangement: regular, square.
—pitch: 7¾ in.
Resonance escape factor: $p = 0.89$.
Fast fission factor: $\epsilon = 1.03$.
Multiplication factor: $k_\infty = 1.054$ hot poisoned.
Burn-up, individual elements: 4,500 MWd/t or 5 years max.
Method of flattening: steel bars in selection of 224 special holes.
Neutron flux (maximum): 2.0×10^{13} n/cm² sec (Westcott Convention);
3.2×10^{13} n/cm² sec (Story Convention).

PRESSURE VESSEL:
Shape and size: sphere 61 ft i.d.
Thickness of plate: 3½ in.
Material: Si-killed C/Mn steel.
Working pressure: 240 p.s.i.g. at reactor inlet.
Support type: skirt.
No. of standpipes: 241.
Volume ratio: core and reflector/vessel: 41.5%

COOLANT:
Gas: carbon dioxide.
Inlet temperature to reactor: approx. 200°C.
Outlet temperature from reactor: approx. 400°C.
Mass flow: 9,000 lb/sec.
Number of circuits: 6.
Main duct diameter: 5 ft 6 in; thickness: 15/16 in.

CIRCULATORS:
Number: 6.
Type: axial S.R.S.
Drive: squirrel cage induction motor.
Speed: 2,950 rev/min.
Method of flow variation: linked throttle and by-pass valves, former never quite closed.

CONTROL:
Number of channels usable for control rods: 241.
Diameter of control rod channels: 3.5 in.
Total number of rods: up to 185.
Safety rods: 12.
Number of control sectors: 9.
Control rod material: boron steel in S.S. tube.
Rod length: approx. 26 ft.



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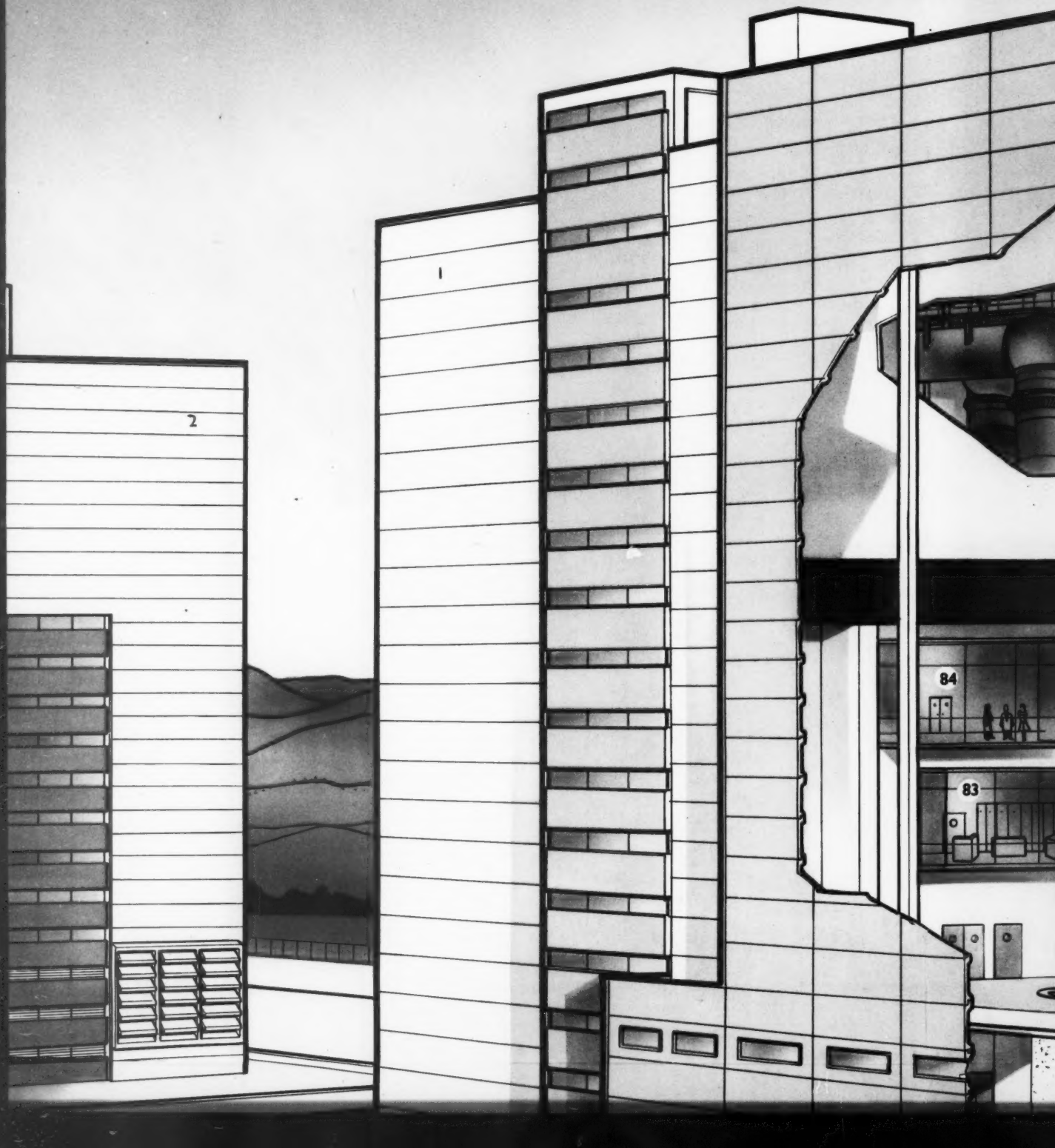

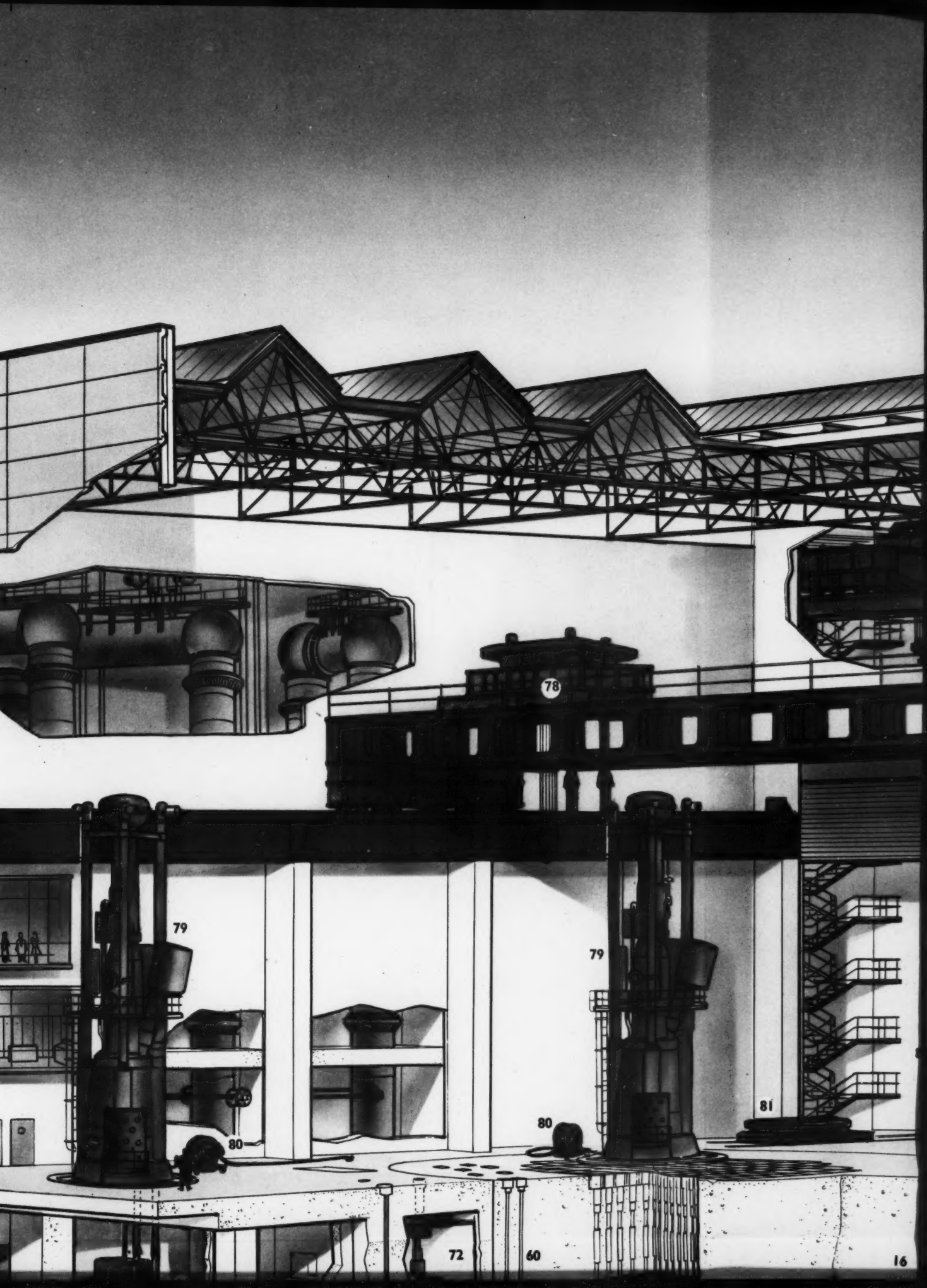
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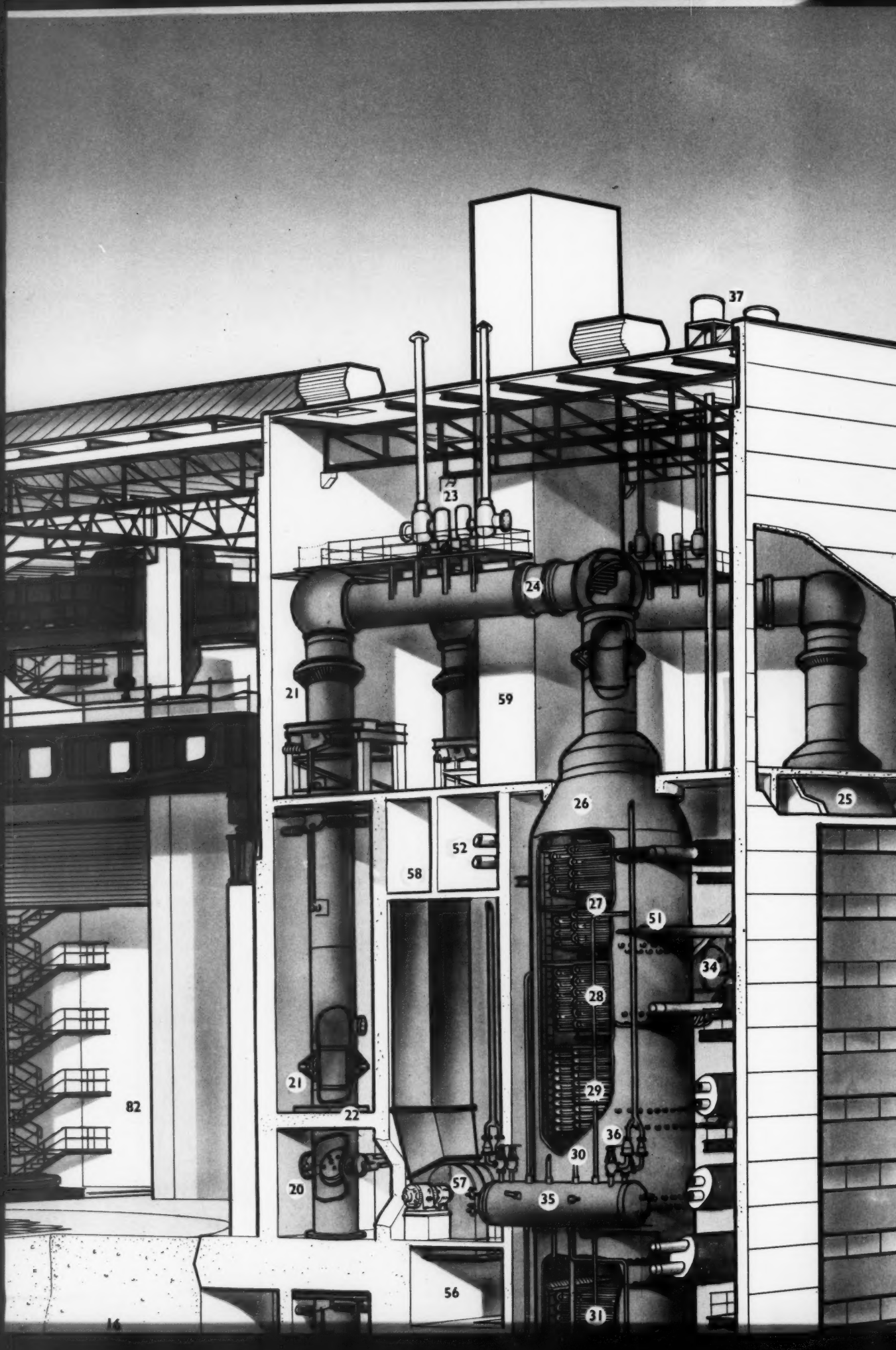
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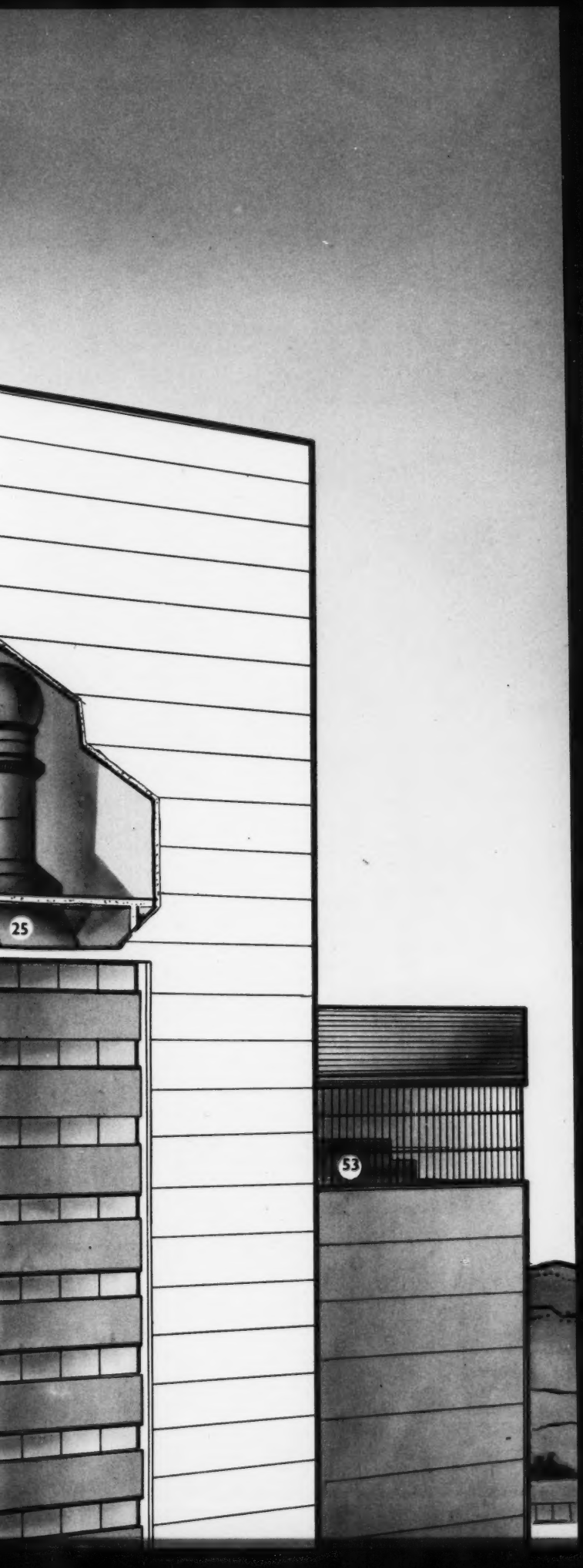
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11% v
Vacuum:

ALTERNATORS:

Rating/p
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Cooling-

DUMP CONDENSERS:

No. and
Capacity

Number of channels usable for control rods: 241.
Diameter of control rod channels: 3.5 in.
Total number of rods: up to 185.
Safety rods: 12.
Number of control sectors: 9.
Control rod material: boron steel in S.S. tube.
Rod length: approx. 26 ft.
Rod diameter: 2¼ in.
Method of suspension of rods: cable.
Method of drive: static-type low frequency a.c.
Automatic control of sector rods: error signal from sector outlet gas temp.
Automatic control of coolant flow: error signal from turbine HP inlet pressure.

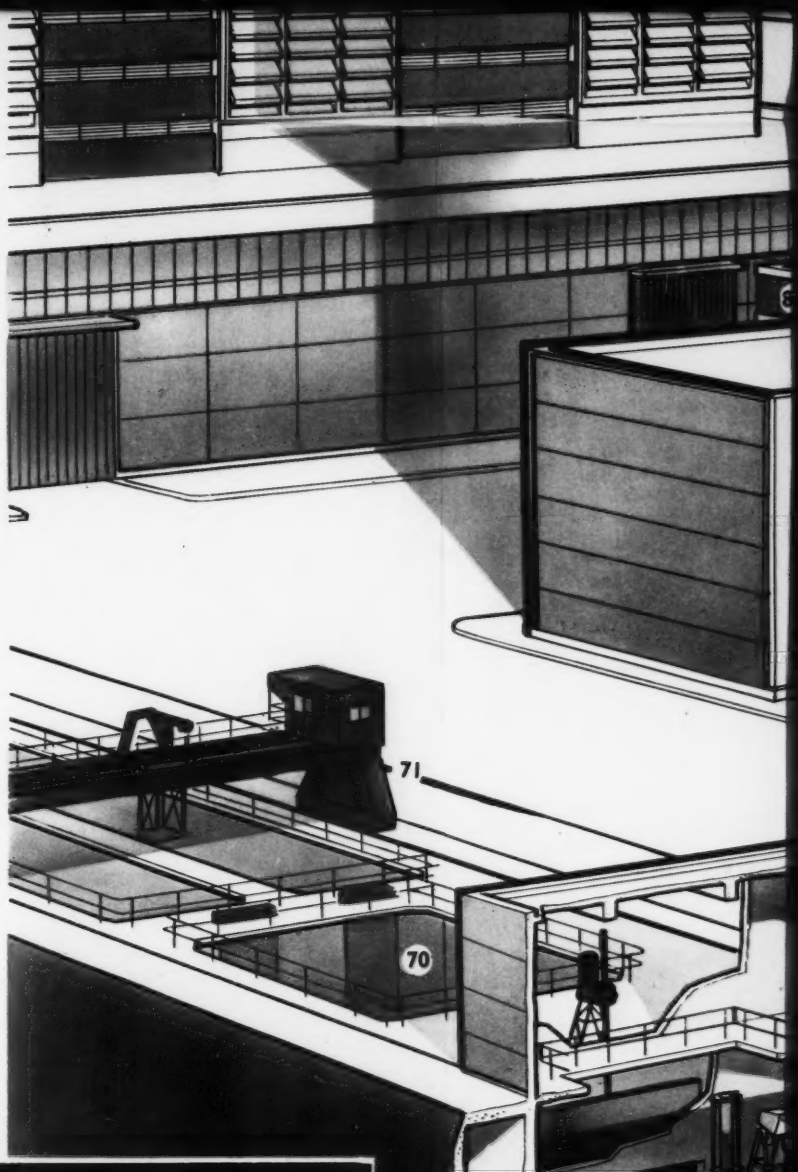
Thermal shield: two ½-in steel plates form air passage.
Biological shield sides, max. thickness: 10 ft 6 in; min. thickness: 10 ft.
Biological shield top thickness: 11 ft 6 in.
Shield cooling: induced draught air.

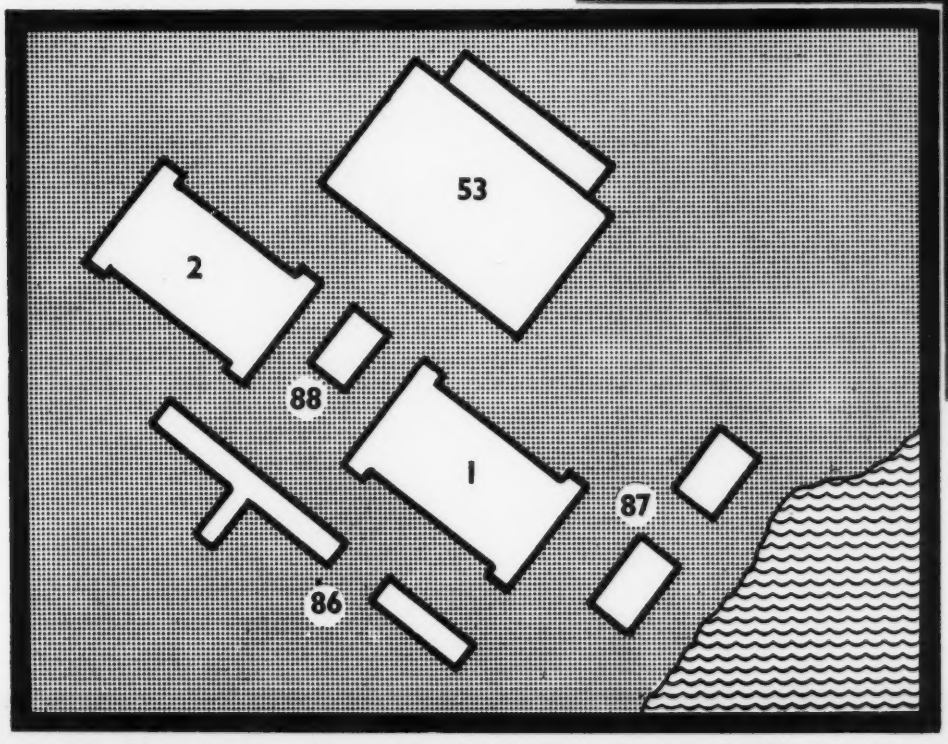
Number per reactor: 6.
Type: shell and tube; forced recirculation.
Main shell diameter: 18 ft; height: 116 ft.
Plate thickness: 2½ in max.
Plate material: boiler quality steel.
Element construction: resistance welded external fins throughout.
Rating: 150 MW heat.
Steam conditions at turbine stop valves:—
HP: Pressure: 927 p.s.i.g.
Temp.: 715°F.
Flow ratio: 65%.
Flow rate: 290,000 lb/h.
LP: Pressure: 290 p.s.i.g.
Temp.: 685°F.
Flow ratio: 35%.
Flow rate: 115,000 lb/h.
Feed temperature: 226°F.

Number per station: 4.
Construction: four cyls. six exhausts.
Type: one impulse stage, rest reaction.
Maximum continuous rating: 145 MW.
Speed: 3,000 rev/min.
Condensers per turbine: 2.
Conditions at water separator between HP and LP cyls.: 7 p.s.i.a., 11% wet.
Vacuum: 28.85 in Hg.

Rating/power factor/S.C. ratio: 170 MVA/0.85/0.4.
Generation voltage: 16.5 kV.
Cooling—stator winding: water.
—stator iron and rotor: hydrogen 30 p.s.i.g.

No. and type: 2 vacuum.
Capacity: each 10% of reactor heat.





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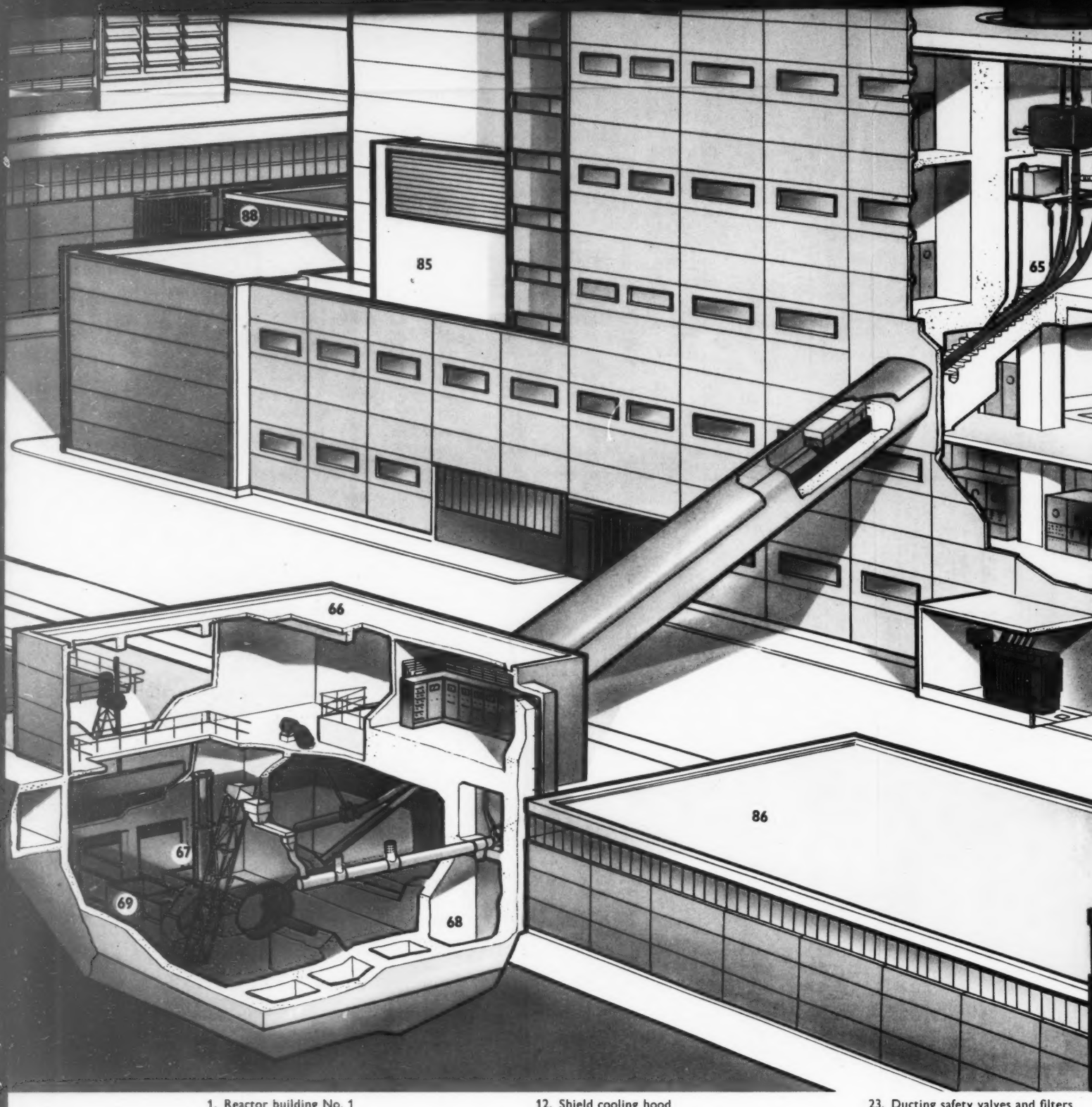
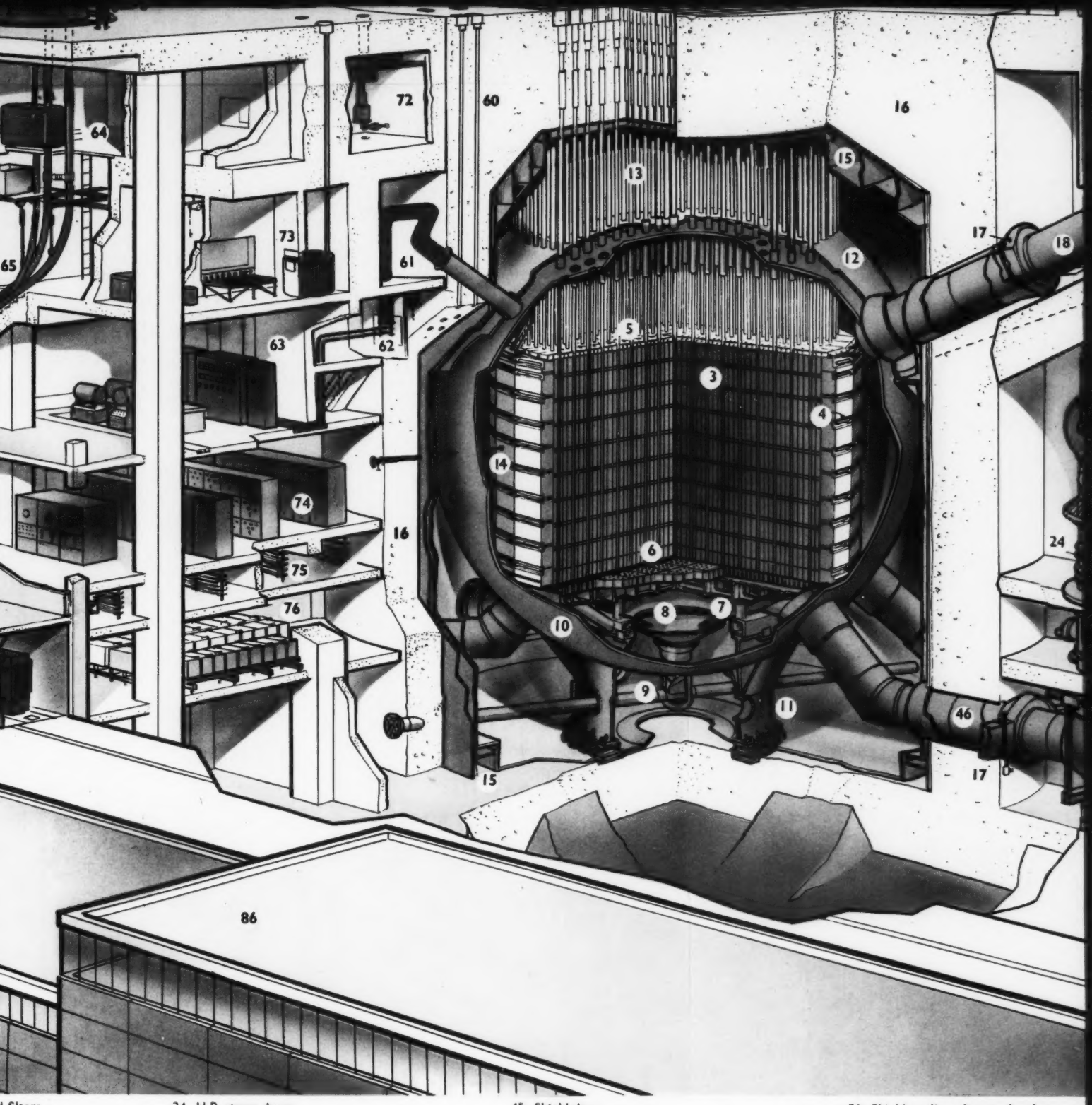


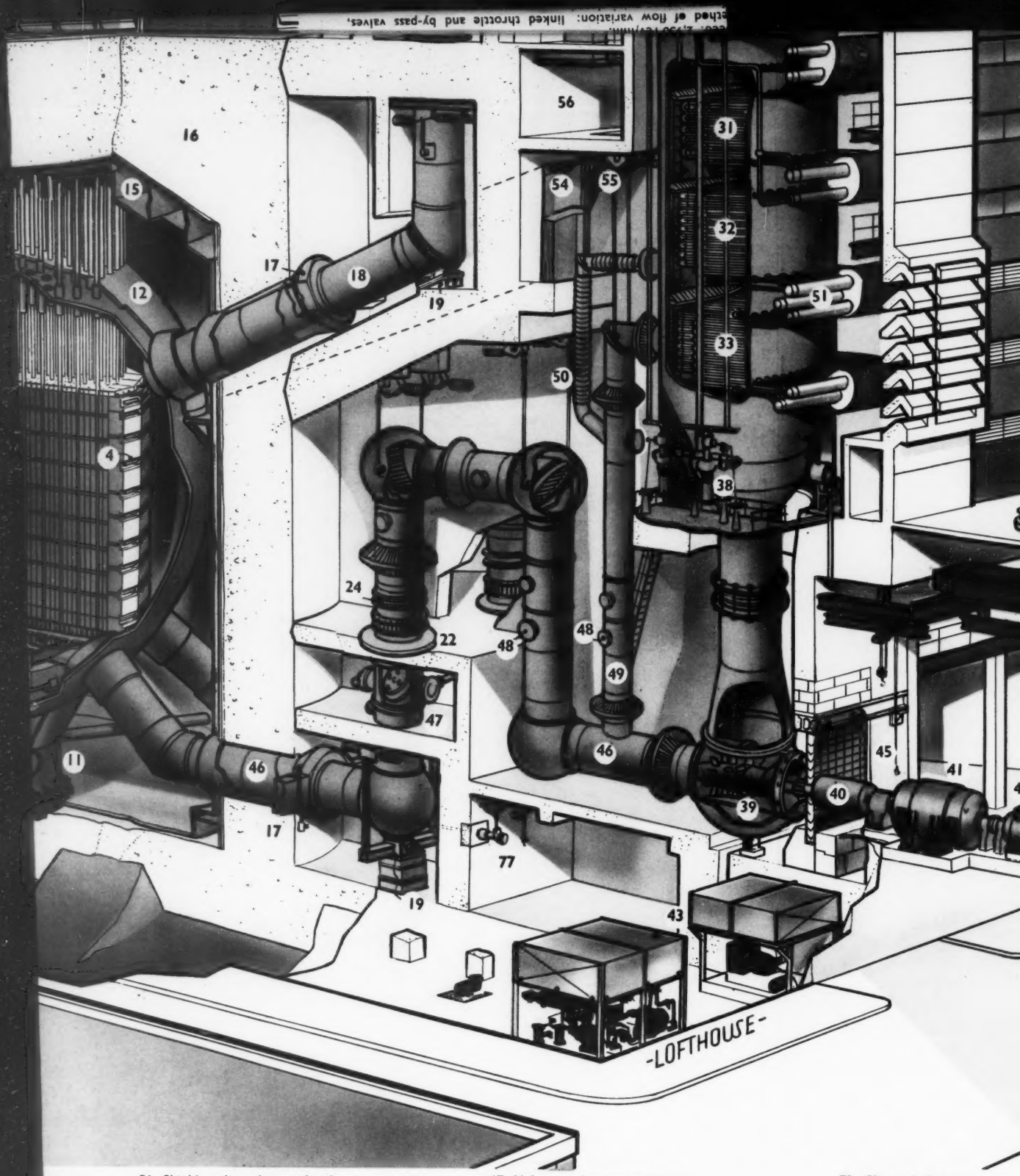
KEY

1. Reactor building No. 1
2. Reactor building No. 2
3. Core
4. Core restraint structure
5. Charge pans
6. Diagrid
7. Core support structure
8. Internal debris fairing
9. Debris removal ducts
10. Pressure vessel
11. Support skirt
12. Shield cooling hood
13. Standpipes
14. Wigner probe
15. Shield cooling ducts
16. Biological shield
17. Shield cooling air inlet sleeves
18. Hot gas duct
19. Hanger failure emergency supports
20. Hot gas duct isolation valve
21. Bellows unit
22. Containment seals
23. Ducting safety valves and filters
24. Removable spool piece
25. No. 1 steam raising unit
26. No. 2 steam raising unit
27. H.P. superheater
28. L.P. superheater
29. H.P. evaporator
30. L.P. evaporator
31. H.P. economizer
32. L.P. economizer
33. Mixed economizer

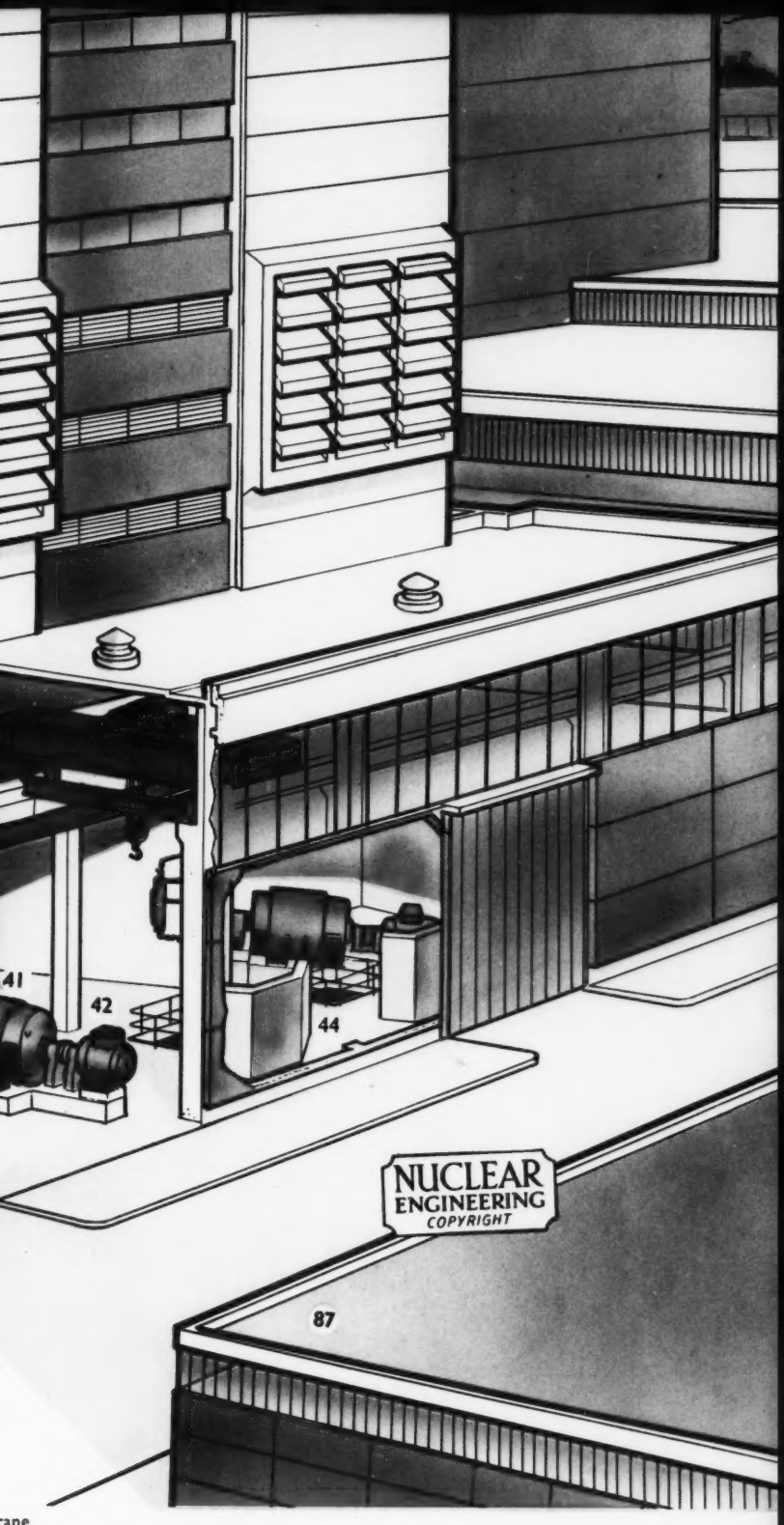


d filters

34. H.P. steam drum
35. L.P. steam drum
36. L.P. drum safety valves
37. L.P. drum safety valve silencers
38. L.P. recirculating pumps
39. Gas circulators
40. Flywheel
41. Circulator main motors
42. Circulator pony motors
43. Circulator lub. oil plant
44. Control panels
45. Shield door
46. Cooled gas duct
47. Cooled gas duct isolation valve
48. Throttle valves
49. Recirculating duct
50. By-pass duct
51. Headers
52. Steam pipe race to turbine house
53. Turbine house
54. Shield cooling extract duct
55. Shield cooling air filters
56. Shield cooling plenum chamber
57. Shield cooling fans
58. Shield cooling discharge duct
59. Shield cooling stack
60. Storage tubes
61. Burst can detection pipes
62. B.C.D. selector valve
63. B.C.D. room
64. Irradiated fuel disposal room
65. I.F.D. chute
66. I.F.D. acceptance bay



56. Shield cooling plenum chamber
57. Shield cooling fans
58. Shield cooling discharge duct
59. Shield cooling stack
60. Storage tubes
61. Burst can detection pipes
62. B.C.D. selector valve
63. B.C.D. room
64. Irradiated fuel disposal room
65. I.F.D. chute
66. I.F.D. acceptance bay
67. Volume reduction equipment
68. Magnox storage vaults
69. Irradiated element skips
70. Cooling pond
71. Skip handling crane
72. Active handling cell
73. New fuel magazine
74. Essential services room
75. Cable race
76. Battery room
77. Debris probe
78. Charge hall crane
79. Fuelling machines
80. Fuelling machine service t
81. Fuelling machine transpor
82. Fuelling machine maintena
83. Charge face control room
84. Visitors viewing room
85. Shield cooling air inlet filt
86. Ancillary buildings
87. New fuel store
88. Main change room



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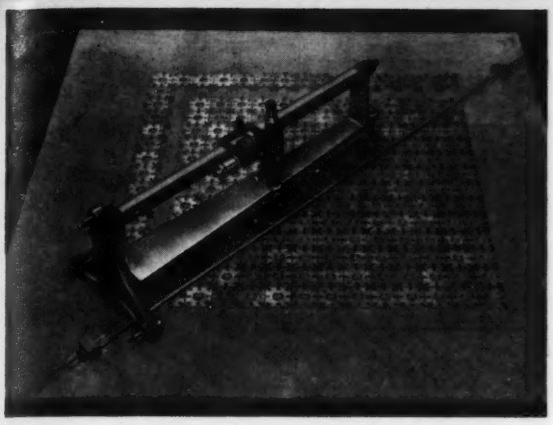


Fig. 3.—Model to investigate the properties of a tile layer.

in the block and tile faces. Clearances are chosen to ensure that the locating spigots carry all mechanical loads during erection and operation, and the seals only carry the differential pressure. During operation they are allowed to deflect radially on to the outer faces of the grooves. In spite of this precaution some leakage remains and, together with the loss of coolant, there results a pressure at the bottom of the control holes, which is transmitted from the flattened zone where it is highest radially outwards through the tile layers and resisted by the garters.

The leakage would be reduced even more if the flow resistances in series with the out of channel resistance could be increased as well. But it is not practical to seal control holes formed in four different columns subject to differential vertical movements. Even if it were, to trap pressure in the gaps surrounding blocks is to create forces tending to disrupt the core as a whole; while radial forces could be contained by increasing the garter loads or raising the gas seal level, upward ones would tend to lift off the top reflector. Careful design study shows in any case that if differential strains are provided for everywhere there are so many potential leak paths that the pressure in these gaps cannot be controlled and the only solution is the one adopted, namely to seal fuel channels.

Differential Strains

Variation of irradiation strains may occur from one block to the next. In the Trawsfynydd core, there is clearance vertically at the tile keys and other places where columns overlap so that adjacent columns may grow or shrink by different amounts without constraint (absolute vertical strain is of no significance). Similarly, while mating keys and keyways are nominally at the same flux and temperature, the choice of keyway clearance provides for differential behaviour.

Nothing has been said about stored energy because once the safety analysis shows that no special steps are needed to raise temperatures, this ceases to be a design problem. By the same token, variations in physical properties are simply allowed for in calculation.

The total distortion of a fuel channel or a control hole would consist of the keyway clearances accumulated across the core, which amount to only a fraction of an inch, and of very small internal steps due to manufacturing tolerances or local differential strains, much smaller than the lead-in on the components that go into it. The structure as a whole is inherently stable, and working faces or high stresses have been avoided. The core is free of internal constraint and independent of exact values of Wigner strain.

Trawsfynydd Contractors, Preliminary List

APC Associated Companies

CROMPTON PARKINSON, LTD. 275-kV generator and station transformers. Unit and auxiliary transformers. 11-kV 750-MVA switchgears. 415-V switchgear. Switchgear relay panels. Auxiliary motors and alternators. Cables, supply and installation. Lighting equipment.

FAIREY ENGINEERING, LTD. Installation of burst can detection equipment. Graphite machining and laying; core components. Charge and discharge equipment. Reactor control equipment. Gas filters. Throttle by-pass valve control system. Charge pans.

INTERNATIONAL COMBUSTION (NUCLEAR APPLICATIONS), LTD. Boilers and ancillary plant. CO_2 ducting. Biological shield cooling system. Miscellaneous fabricated equipment. By-pass filters.

RICHARDSONS WESTGARTH AND CO., LTD. Turbines, alternators, etc. CO_2 circulators. Bellows unit. Graphite levelling plates. CO_2 ducting cascade bends. Extractor ring main.

NUCLEAR CIVIL CONSTRUCTORS. Civil engineering works.

Other Contractors

AITON AND CO., DERBY. Main steam and feed pipework. Pumphouse C.W. pipes and valves. Auxiliary pipework, etc.

SIR WM. ARROLL AND CO., LTD., two 450-ton max. capacity Elec. O/H travelling cranes.

AUTOMATIC TELEPHONE ELECTRIC CO. Alarm annunciator equipment.

BABCOCK AND WILCOX, LTD. Reactor pressure vessels. Goliath crane. Portmadoc jib crane.

BATES BROS. (ENGINEERS), LTD. Design for drop wedge gates. Heating and ventilating plant.

BLACKBURN ELECTRONICS, LTD. Temperature data processing equipment.

J. BLAKEBOROUGH AND SONS, LTD. Main CO_2 control valves. By-pass bleed valves.

BOVING AND CO., LTD. Valves.

F. W. BRACKETT AND CO., LTD. C.W. band screens.

BRITISH INSULATED CALLENDERS CABLES, LTD. Thermocouples.

THE BRITISH REFRASIL CO., LTD. Internal insulation for reactor pressure vessel.

BURNETT AND LEWIS, LTD. CO_2 Adsorption driers.

CONSOLIDATED PNEUMATIC TOOL CO. Compressor equipment.

CONSTRUCTORS JOHN BROWN, LTD. Instrument panels and cubicles.

COWANS SHELDON AND CO., LTD. Turbine house overhead travelling cranes.

THE DARLINGTON INSULATION CO., LTD. Lagging of cold leg ducting.

DAVIDSON AND CO., LTD. Shield cooling fans.

DEWRANCE AND CO., LTD. H.P. and L.P. steam and feed valves.

THE DISTILLERS CO., LTD. Liquid CO_2 storage plant.

DRAKE AND GORHAM (CONTRACTORS), LTD. Lighting and small power installation.

DRYSDALE AND CO., LTD. C.W. pumps.

ENFIELD ROLLING MILLS, LTD. Generator connections.

E.S.I., LTD. Auxiliary pipework.

WM. E. FARRER, LTD. Penstocks. Panel screens. Sewage disposal equipment.

FISHER GOVERNOR CO., LTD. Steam dumping valves.

FOSTER TRANSFORMERS, LTD. Transportable transformers. Transformers.

GLENFIELD AND KENNEDY, LTD. Main C.W. valves. Auxiliary and sluice valves.

GWYNNES PUMPS, LTD. Strained water booster pumps.

HACKBRIDGE AND HEWITTIC ELECTRIC CO., LTD. Mercury arc rectifiers.

G. N. HADEN AND SONS, LTD. Fixed oil handling and filtration plant.

HAYWOOD TYLER AND CO., LTD. Boiler circulator pumps.

HOPKINSONS, LTD. Main CO_2 valves. Boiler mountings.

LANCASHIRE DYNAMO NEVELIN. Control rod contactor boards.

ALBERT MANN ENGINEERING CO., LTD. Throttle by-pass valve control shafts, etc.

MATHER AND PLATT, LTD. Motors for CO_2 circulator drives. Water jet fire-fighting equipment. Emergency boiler feed pump.

MUIRHEAD AND CO. Rod position indicator.

MB METALS, LTD. Connector for fuel element temperature thermocouples.

NECKAR WATER SOFTENER CO., LTD. Filters for site temp. water supply. Station water treatment plant.

PATERSON HUGHES ENG. CO., LTD. Skip handling machine. Cooling pond dispatch area crane. C.W. pump house crane.

BRUCE PEEBLES AND CO. Control rod low frequency generators. Motors for boiler feed pumps.

PLESSEY NUCLEONICS, LTD. B.C.D. precipitators. B.C.D. selector valves.

PRITCHETT AND GOLD AND E.P.S. Batteries and chargers.

PYRENE CO. Dry chemical fire fighting equipment.

REDHEUGH IRON AND STEEL. C.W. pipes.

REDPATH BROWN, LTD. Turbine house and reactor building structural steelwork.

RUSTON AND HORNSBY, LTD. Couplings. Butt weld end connections.

RUSTON PAXMAN SALES DIVISION. Diesel alternator sets.

B. SAXTON AND CO., LTD. Thermocouple troughing.

R. W. SCOTT AND SONS, LTD. Anchor pipes and plates.

STAHL-ARMATUREN. CO_2 valves.

STANDARD TELEPHONE, LTD. P.A.X. exchange equipment.

STANTON IRONWORKS. Pipes and fittings.

STEWART AND LLOYDS, LTD. Reactor disposal chutes.

SULZER BROS. (LONDON), LTD. Boiler feed pumps.

THAIN, ROBERTSON, LTD. Extraction fans.

TUBE PRODUCTS, LTD. Finned tubing.

ULTRA ELECTRONICS, LTD. Sector servo system.

UNITED KINGDOM ATOMIC ENERGY AUTHORITY. Nuclear graphite.

WALLACE AND TIERNAN, LTD. Chlorinating plant.

ROBERT WATSON AND CO. Single-span cone roof.

ALLEN WEST AND CO., LTD. Switchfuse and contactor equipment.

THE WHITEHEAD IRON AND STEEL CO., LTD. Reinforcing steel bars.

YORKSHIRE IMPERIAL METALS, LTD. Copper tubing and fittings.

JAMES YOUNG AND CUNNINGHAM. Cast-iron valves, bends, flanges, tees, etc.

Industrial Graphite Impregnation

FACILITIES for giving graphite components permeabilities to the order of 10^{-6}cm²/sec are now available in the United Kingdom on an industrial scale. The Hawker Siddeley Nuclear Power Company have put into operation at their Langley, Slough, works an impregnation plant which will be used to treat graphite for a wide variety of applications, including the fuel tubes and boxes for the Dragon project at Winfrith (see *Nuclear Engineering*, July, 1960, p. 314).

The method of impregnation used is derived from the furfuryl alcohol process developed at the Royal Aircraft Establishment, Farnborough. The company have evolved some refinements of their own and introduced some new impregnants. With "standard" treatment the degree of impermeability falls off from the surface inwards which implies that machining should not take place after impregnating, but where impermeability is required throughout a component a special treatment will achieve this.

The effect impregnation has on some of the physical and mechanical properties of graphite is shown in Table 1. Except where otherwise stated, these measurements were made at room temperature under ordinary atmospheric conditions. Some preliminary measurements on flexural strength at elevated temperatures have been made and found satisfactory. It is expected that similar improvements will occur in tests for elasticity at the same temperatures.

TABLE I.—Some Mechanical and Physical Properties of Impregnated H.S. Graphite

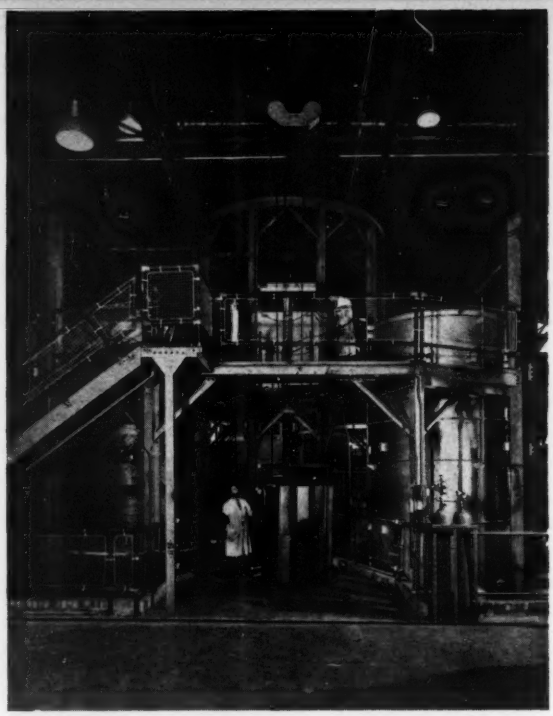
	Untreated	Treated
Flexural Strength, lb/in²	3,600	5,700
Elastic Modulus, lb/in²	1.8×10^6	3.0×10^6
Electrical Resistivity, mΩcm	1.9	1.5
Thermal Expansion/°C at 300° C	2.0×10^{-6}	2.1×10^{-6}
Thermal Expansion/°C at 600° C	2.2×10^{-6}	2.7×10^{-6}
Thermal Expansion/°C at 900° C	2.6×10^{-6}	3.0×10^{-6}
Ultimate Tensile Strength, lb/in²	3,200	5,900

A notable omission from the table of properties is that of thermal conductivity. The reason, apparently, is that it had not at the time of the compilation of the table been possible to obtain a series of highly consistent measurements. Preliminary results do indicate, however, that improvements will be recorded.

Oxidation resistance of the impregnated graphite is said to be improved; there is, however, no information on the effects of radiation. It is understood that relevant studies are being made by the U.K. AEA and the Dragon development team.

Impregnation Plant

The carbonizing furnace, comprising a cylindrical three-zone heating chamber, 4½ ft dia by 14½ ft deep, enclosed in a 22 ft high by 8½ ft dia structure, operates up to temperatures of 1,000° C and is designed for operation with inert gases. At present, nitrogen having a maximum oxygen content of 10 ppm is used.



The impregnation plant at Langley. The carbonizing furnace is on the right and the graphitizing furnace on the left. Controls for both furnaces are on the platform behind the 3-ton charging gantry.

Operating temperatures are much higher in the graphitizing furnace—for some applications the graphite has to be heated up to 3,000° C, though in the case of the Dragon components a temperature of 1,700° C in a vacuum of 0.1 mm Hg may be sufficient. The furnace which was designed by Spembly, Ltd. (the main contractors for the whole plant being Sintering and Brazing Furnaces, Ltd.) was built to take components 10 ft long requiring a power supply of 800 kVA. Heating within the steel pressure vessel is accomplished by an annulus of 96 graphite rods 6 ft long which sit on 12 junction boxes connected to bus bars beneath.

In marked contrast to the complex and sizeable furnaces are the simple but important impregnation facilities. The method of impregnation used in any specific case depends upon the size and shape of the component. Hollow components are filled up with resin and a pressure applied. They are considered to be impregnated when the resin starts to seep through. The time this takes depends on the quality of the graphite and previous treatment. Smaller components are impregnated by placing them in a receptacle which is evacuated and then filled with resin. They are left to soak for several hours.

Methods of Testing

Equally unspectacular, and yet as important, are the facilities for testing the permeability of the graphite after it has cooled down from the graphitizing furnace. Two testing methods are used. For measuring permeabilities in the 10^{-2} to 10^{-5}cm²/sec range, one side of the component is subjected to a known pressure of gas, and the time taken for this to drop over a predetermined level recorded.

In the case of lower permeabilities, the rate of a vacuum decay over a set period is measured.

To measure distribution of permeability, destructive testing has to be used.

It is estimated that the new plant at Langley is capable of handling about 100 ton of graphite a year. Later it is planned to install three more carbonizing furnaces and another graphitizing one.

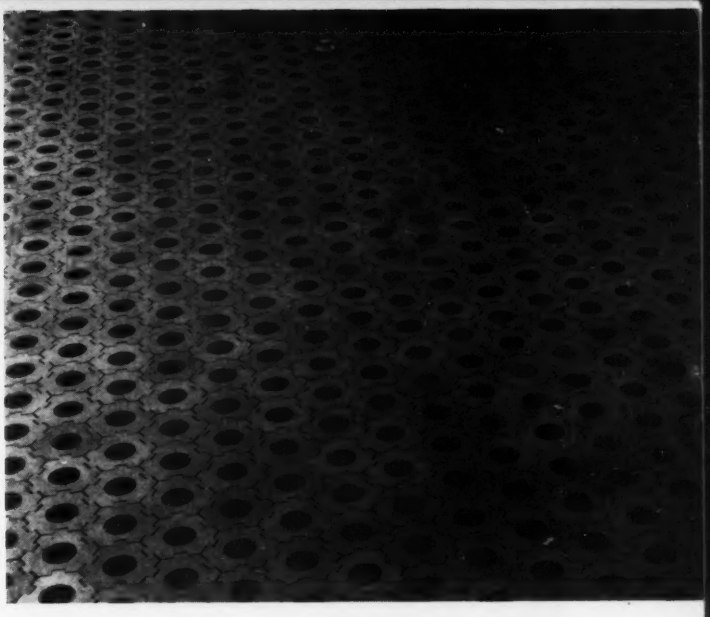


Fig. 1.—Model illustrating the non-compacted design adopted for the Tokai-Mura reactor.

Earthquake-resistant Core for Tokai-Mura

by R. W. BAILEY, B.E., B.Sc., Ph.D.
and J. L. HEAD, B.Sc.(Eng.)
(*Structural Development Department, G.E.C. and Simon-Carves Atomic Energy Group, Erith*)

JAPAN'S first nuclear power station, designed for the Japan Atomic Power Company by the G.E.C. and Simon-Carves Atomic Energy Group, is of the Magnox type. Japan is in an earthquake zone and, although the Tokai-Mura site, 70 miles north-east of Tokyo, has been carefully selected for its good history in this respect, earthquakes are still possible. The major problem associated with this design has, therefore, been the development of a graphite core able to resist the additional loads occurring during an earthquake.

A difficulty in seismic design is that these additional loads cannot be predicted exactly, and will depend on such factors as the type of structure and the foundation and damping characteristics. In 1923 it was recognized in Japan that it is the horizontal accelerations occurring during an earthquake that cause most of the damage. It was proposed that structures should be designed to withstand a horizontal force at any level equal to the weight of the structure above that level multiplied by a "seismic coefficient," the magnitude of which was left to the individual authority concerned. This method for taking into account earthquake loading has been incorporated into the various building codes of countries in earthquake zones.

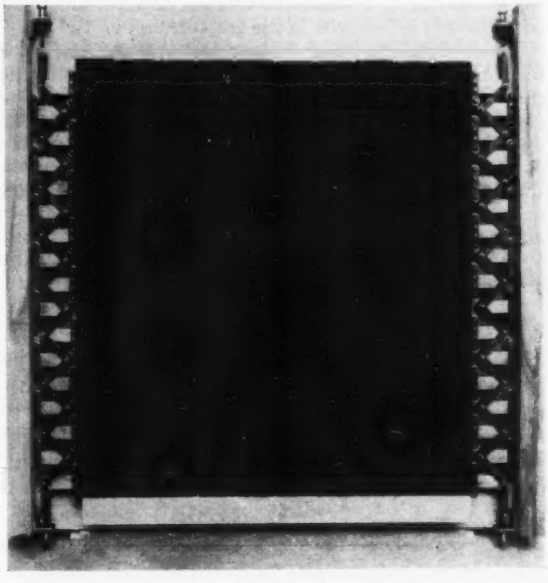


Fig. 2.—A compacted type of core design is shown in this model.

In 1950 the current Japanese Building Code was drawn up, specifying a seismic coefficient of 0.2 with a provision for increasing this factor by 0.01 for each 4 m (13 ft) by which the height of the structure exceeds 16 m (52 ft) above ground level.

The purpose of the code is to ensure that buildings will have adequate strength to limit, although not necessarily to prevent, structural damage during a strong earthquake. In the case of a reactor core structure, because of inaccessibility for repair once the reactor has been commissioned, it is essential that no damage should occur that would prevent the functioning of the reactor both during and after an earthquake. Consequently, in the design of the core structure for this reactor a seismic coefficent of three times that given by the Japanese Building Code was specified.

This would appear to be a very arbitrary design criterion. However, dynamic studies carried out in Japan, and studies of historical records of the magnitude of earthquakes and resulting damage to buildings that have occurred in this particular locality, have shown that these assumptions do form a reasonable basis for design.

In the design of the core structure, the problem has been to provide for the transmission of these horizontal forces throughout the core, while at the same time allowing free movement of the core to accommodate thermal and irradiation effects. The magnitude of this problem will be appreciated when it is realized that the weight of the core structure is approximately 2,000 ton and that the total horizontal force which the core structure must be designed to carry is of the order of 1,300 ton.

Original Core Proposals

The core was originally designed at a time when the available data on the dimensional changes that take place in graphite under irradiation were extremely limited. The information available indicated that graphite would grow under irradiation, and extrapolation of the data indicated that this growth would continue throughout the life of the reactor. The proposed core was in the form of a cylinder, made up of columns of graphite of square section, each column consisting of alternate "bricks" and "tiles." The columns were arranged so that the tiles would butt together on all four faces to form continuous layers across the core, whilst the blocks were spigoted into the tiles. The tile layers effectively stabilized the blocks against lateral forces arising during an earthquake, and gaps left between block faces would allow the anticipated Wigner growth to take place freely. By using pressed graphite for the tiles, so that the horizontal plane corresponded to the direction of

minimum growth rate, the overall growth of the core would be minimized, thus simplifying the design of the restraint structure.

This restraint structure consisted essentially of a cylindrical network of columns with helical bracing. By using stainless steel columns and mild-steel bracing it was possible to choose a helix angle such that the radial thermal expansions of the restraint structure and of the tile layers were identical. A prestress was applied to the structure, which would then transmit the earthquake loading to the supporting grillage by means of the helical bracing. Irradiation growth of the tile layers could be accommodated by elastic deformation of the structure.

Having based the design of the core on the assumption that graphite would grow under irradiation, further data then became available which indicated that this growth would not continue throughout the life of the reactor. It became apparent that after a small initial growth the graphite would then begin to shrink and that this shrinkage would continue throughout the remainder of the reactor's operational life.

Application of this data to the proposed core design indicated that the prestress between the graphite and the restraint structure would be lost, and substantial gaps opened between the graphite tiles such that the structure could become unstable during an earthquake.

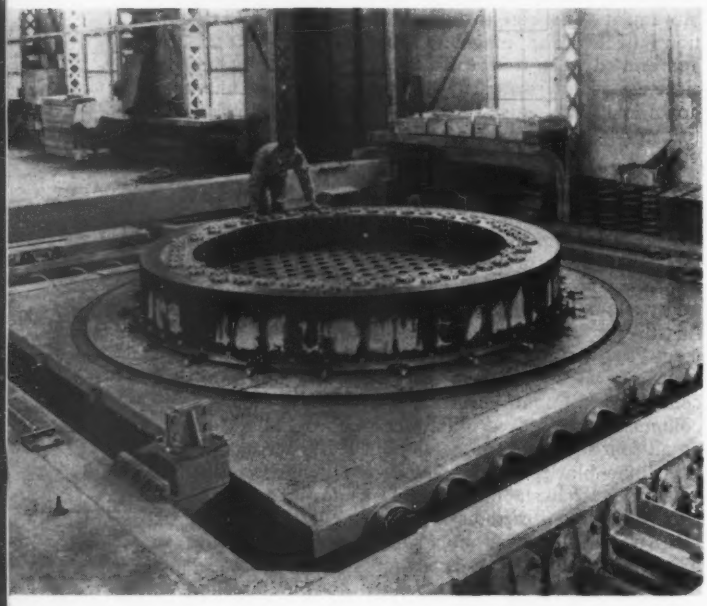
CORE DESIGN DEVELOPMENT

Exhaustive studies were carried out on a number of alternative core designs, in each of which stability is maintained despite shrinkage of the graphite. The designs that were considered can be grouped into two distinct types.

Compacted Core Designs

The principle of these designs is that as shrinkage of the graphite takes place the core is compacted by the application of radial forces at the boundaries. In this way gaps are not permitted to form, and the horizontal loads on the core during an earthquake can be transmitted to the restraint structure by the induction of compressive forces between bricks. The model illustrated in Fig. 2 shows an example of this type of design. No tile layers are used, and the temperature-compensated restraint structure is replaced by a mild-steel cylindrical shell, stiffened by circumferential rings. Between this cylinder and the graphite core are placed tightening linkages, uniformly spaced around the circumference and operated from outside the biological shield. During start-up and shut-down of the reactor the linkages are operated automatically to allow for different thermal expansion and contraction rates between graphite and steel. Further adjustment is necessary at intervals throughout the life of the reactor to make allowance for the shrinkage of the graphite. Stability is an important feature in the design of the mechanism and the tightening must be carried out in a preselected sequence.

Fig. 3.—Single-layer model for studying boundary reaction under dynamic loading. *(By courtesy of the Building Research Institute, Japan.)*



Non-compacted Core Design

In this type of design, gaps are permitted to open up within the core due to Wigner shrinkage and thermal effects. The model illustrated in Fig. 1 is an example of this type and is the design that was finally chosen for the J.A.P.C. reactor.

In this design the core is made up of layers of blocks, each block being hexagonal in cross-section so that the fuel channels are arranged on a triangular lattice. Within the body of the core each block is located relative to the surrounding six blocks by six keys running the full length of the block and arranged radially to it. The boundary blocks are attached directly to the restraint structure, which consists of a reinforced cylindrical shell, so that every block is then effectively fixed in plan, relative to the restraint structure. The blocks are staggered between layers to maintain alignment of the fuel channels.

The blocks are machined with integral keys and have keys and keyways on alternate faces. Clearances are specified such that the transmission of load between blocks can only be by induced shear forces in the keys. Horizontal earthquake loads are transmitted in this manner to the boundary blocks and finally to the restraint cylinder. However, for uniform radial displacements at the boundary such as occur due to differential temperature effects, each layer of the core will operate as a mechanism, uniform gaps will be opened up between the faces of adjacent blocks but no loads will be induced in the graphite.

To allow for the initial Wigner growth, small clearances will be left between the faces of adjacent blocks when the core is assembled. Wigner growth will then result in an expansion of each block radially outwards partially to fill these gaps. Subsequent shrinkage will result in contraction of each block on to its own centre. Neither growth nor shrinkage will affect the initial fit between keys and keyways; thus dimensional changes will have no effect on the proper functioning of the core and will only be limited by the initial gaps between block faces and by the dimensions of the keys.

EXPERIMENTAL INVESTIGATIONS

Extensive experimental programmes are being carried out in the laboratories of the Group at Erith, and also at the Building Research Institute of Japan, in order to prove the selected design. The work being done can be classified under several headings, each experiment being designed to prove a particular aspect of the core behaviour.

Vibration Tests on a Single Layer

These tests, carried out at the Building Research Institute, were designed to investigate the circumferential distribution of loading between the core and the restraint structure.

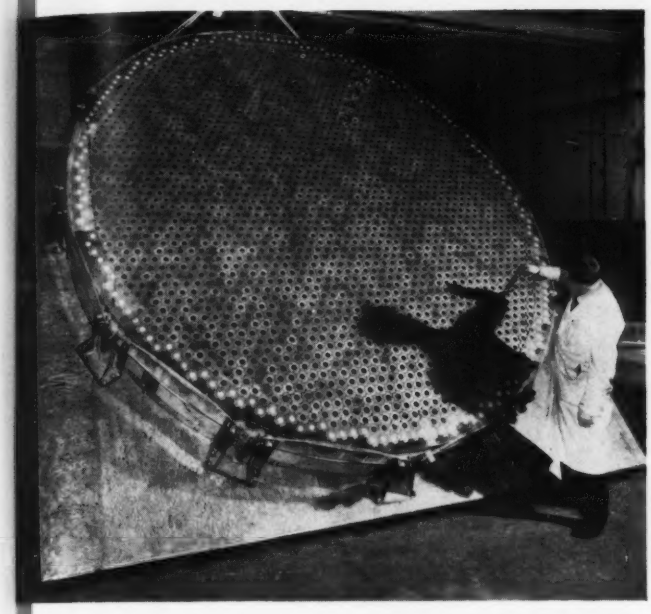


Fig. 4.—Single-layer model for investigations into core deformation and load distribution within the core under lateral loading.

The model, shown in Fig. 3, consisted of a single layer of graphite bricks, 9 ft in diameter, each brick being in cross-section an approximately full-scale representation of the actual brick. The model was attached at the boundary to a mild-steel ring, by means of cantilevers in a manner similar to that to be used in the actual core. The model was tested on a shaking table in order to simulate earthquake loading.

By strain-gauging the cantilevers the interaction between the core and the boundary was studied. Tests were carried out using cantilevers of different stiffnesses in order to assess the influence of the rigidity of the attachment of the core to the restraint structure on the boundary reaction.

The model used, shown in Fig. 4, was similar to that used in the vibration tests, but the lateral loading was applied by simply tilting the model. The model contained the same number of bricks as a layer of the actual core, but the bricks were only about one-quarter full size. The restraint ring was of Perspex and was designed to simulate the restraint structure.

This model was used to study the deformation of the core under loading and to measure the load distribution within the core. The load distribution was obtained by incorporating in the core a number of special blocks machined with separate keys. The load on each of these keys was assessed by measuring the force necessary to withdraw a key with the model in the tilted position.

This force was proportional to the shear force being transmitted by the key.

Thermal Cycling Test

The same assembly of quarter-scale graphite blocks was used for the test, the object being to demonstrate that the opening and closing of the core due to thermal expansion of the restraint structure would take place freely without any jamming of the keys in keyways that might cause failure of the graphite. In this model the restraint structure was represented by a conical shell, to which the boundary blocks were attached by means of rollers running in rails fixed to the inside face of the cone. When moved vertically the cone imposes a radial movement on the boundary bricks. Actuation of the cone in the vertical direction was carried out automatically between set limits by hydraulic jacks, as shown in Fig. 5. This simulated thermal cycling was repeated many times without any observable jamming within the block assembly.

Three-dimensional Model

The model studies previously described have been concerned with the properties of a single layer of the core. In order to study the axial distribution of load between the core and the restraint structure a three-dimensional model of the core is being constructed.

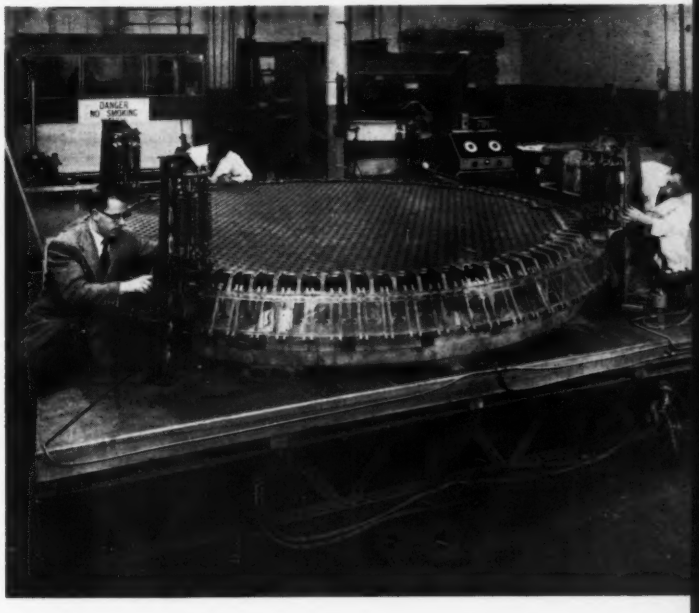
This model consists of six layers of quarter-scale graphite blocks. The restraint cylinder of the actual reactor is replaced by a diagonally braced structure of aluminium, which allows the load coming on to the restraint structure from each layer of the core to be measured by strain-gauging the diagonals. A shaking table has been specially constructed for this test so that measurements can be made under conditions of dynamic loading. The model, which weighs nearly 2 ton, can be subjected to a lateral acceleration of the order of 4 g.

Although the main purpose of this model is to enable a study to be made of the axial distribution of load between the core and the restraint structure, an attempt will be made with this model also to measure the load distribution within a layer of the core under dynamic loading conditions.

Strength Tests on Individual Blocks

The experimental work described above has been designed mainly to study the load distribution in the core assembly under operating and earthquake conditions. In order to assess the safety factor inherent in the design, strength tests are being carried out on graphite blocks having the same cross-section as in the actual core. A method of loading has been developed which produces the forces acting on a block in the core during an earthquake, that is, by the application of shear forces through the keys. Tests of this nature are being used in the optimization of the block profile.

Fig. 5.—Single-layer model designed to demonstrate the effect of thermal cycling.



EGCR— Descendant of Calder Hall

by WILLIAM F. BANKS
(Project Manager, EGCR, Allis-Chalmers Mfg. Co.)

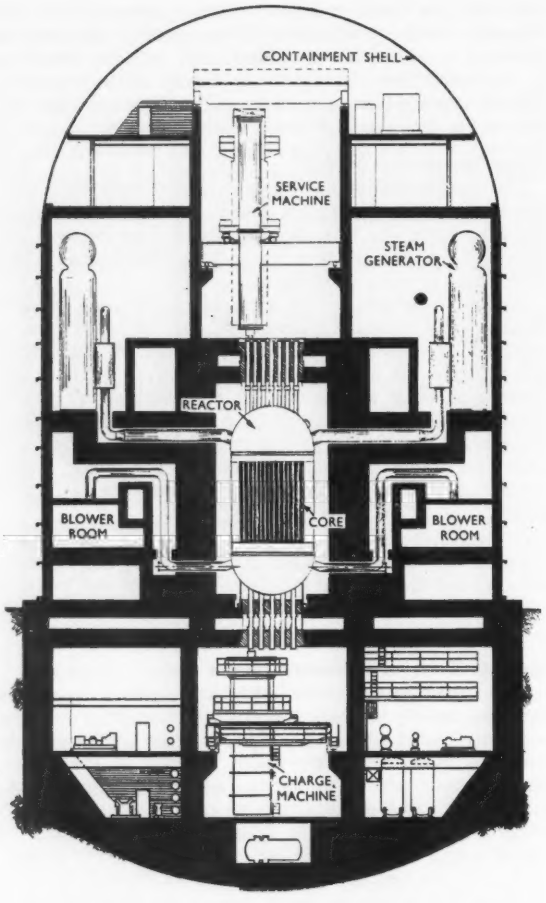


Fig. 1.—General arrangement of the Allis-Chalmers EGCR.

THE Experimental Gas Cooled Reactor (EGCR), which was designed by Allis-Chalmers and is now under construction by the U.S. AEC at Oak Ridge, is a natural outgrowth of the Calder Hall reactors, and represents one of several promising directions of development permitting higher temperatures and smaller size. The ECGR and the Advanced Graphite Reactor (AGR) under construction at Windscale in the U.K. will both use clustered rods of enriched uranium dioxide fuel contained in metal tubes. They both also retain the heterogeneous graphite core, but they differ in a number of secondary but important respects.

One important difference is the use of helium in the EGCR as against carbon dioxide in the AGR. Another difference is the full reliance upon stainless steel cladding in the EGCR, while there is a strong British emphasis upon beryllium, although about 80% of the first AGR core will be stainless steel. Other significant differences relate to the vessel and piping arrangements.

The EGCR type of reactor may be defined as one of the two advanced descendants of Calder Hall which is characterized by the following:

(1) Helium as the coolant.
(2) Ceramic fuel in a pressure-tight stainless steel cladding.
(3) Heterogeneous graphite core.
(4) On-stream replacement of failed elements.
(5) On-stream replacement of exhausted fuel at the end of its life.

In addition to its purpose as a power demonstration plant, the EGCR also will serve as a test reactor. Provision is made for eight experimental loops through the core, four of which will be installed at startup.

Helium cooling

The general problems of helium cooling have been discussed elsewhere by many authors; no detailed discussion of these problems is appropriate here, but there are two general points of interest. On the general question of which gas is most desirable as coolant, hydrogen is outstanding as an agent of heat transfer and heat transport, but because of problems of materials compatibility and safety, it is not usually considered for immediate use. If the principal criterion is power cost, the range of choice quickly narrows to carbon dioxide and helium, both of which offer better thermal performance than nitrogen.

Helium has better *heat transfer* properties than carbon

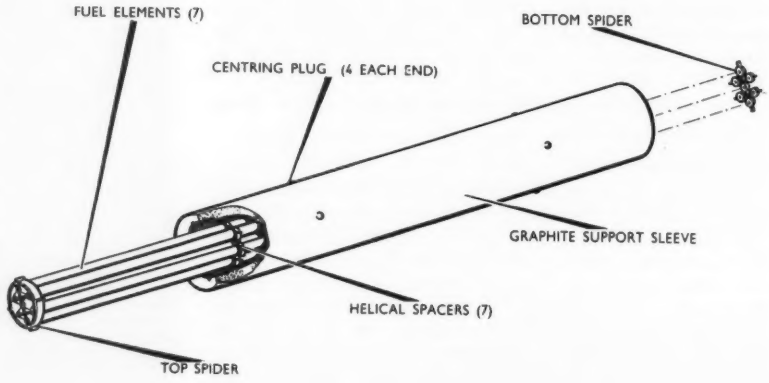
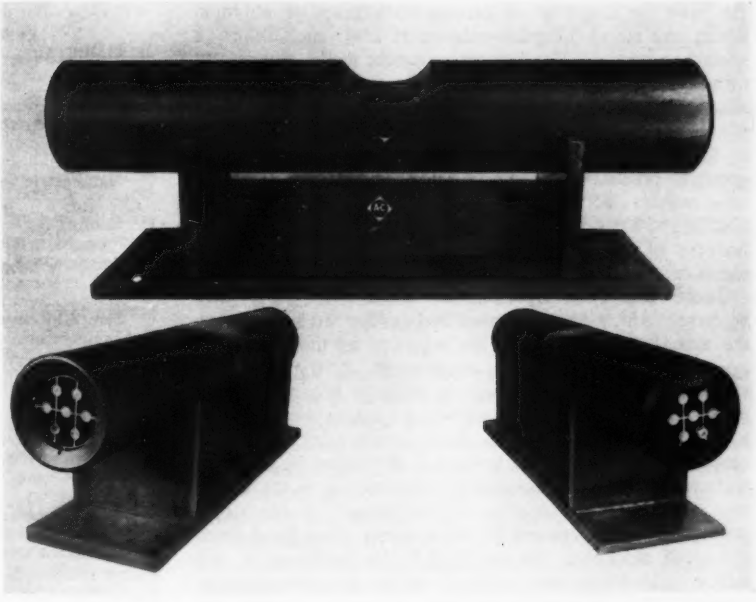


Fig. 2.—Schematic of fuel assembly for EGCR.

Fig. 3.—EGCR fuel assembly.



dioxide, but carbon dioxide has the better *heat transport* capability. The relative importance of these properties depends upon the details of the system under study. Conflicting conclusions may be reached by investigators of similar systems and the two gases appear to be economically comparable. Anticipating future development to the higher temperatures at which carbon dioxide becomes very active chemically, helium appears the better long-range choice.

On the other hand the adoption of helium for the EGCR does not imply complacency as to the ease of solving the related practical problems; only recognition and acceptance of the tasks ahead. To obtain economic power, high temperatures must be attained; to attain high temperatures we must learn to work with helium, and the time to begin this learning is now in order that the difficulties can be evaluated and then resolved.

EGCR Fuel Assembly

The fuel assembly that was designed by Allis-Chalmers for the EGCR is shown in Figs. 2 and 3. It is seen that a graphite sleeve acts as the basic structural member to support the fuel elements individually from a stainless steel spider at the top. A similar spider at the bottom positions the lower end of the elements, while a helical spacer at the midpoint prevents contact between elements. The graphite sleeve is 5 in OD and 3 in ID by 29 in long.

Fuel elements consist of annular uranium dioxide pellets, 0.707 in OD, 0.323 in ID by 0.750 in long, contained within type 304 stainless steel tubes 0.75 in OD, 0.020 in thick. The active length of fuel in each tube is 26.625 in and its density is 10.4 g/cm^3. Six fuel assemblies, plus two dummy assemblies top and bottom, are stacked vertically in each of the 216 active fuel channels of the core.

A total of 427,000 lb/h of helium enters the channels at 313.5 p.s.i.a., 510° F and leaves at 303.25 p.s.i.a. and an average of 1,050° F. The total system pressure loss is 16.5 p.s.i.

An important consideration in the design of this fuel assembly is the flexibility which is provided by the graphite sleeve. Fuel elements of totally different type and configuration can be used in future core loadings, so that the performance of the reactor can be continually improved during its life, without the expensive type of conversion that is required in some other reactors.

EGCR Core

A model of the EGCR core is shown in Fig. 4. The core consists of 120 monolithic graphite columns, 19 ft long. A standard column is 16 in square and contains four fuel channels, 5¼ in ID on an 8-in square pitch. The columns are positioned at their lower ends upon a 3-in plate of carbon steel by square keys. The upper ends of the columns are positioned by a type 304 stainless steel grid which fits between the ends of the columns.

A second grid, above the graphite, supports and positions steel tubes which are extensions of the fuel channels. These steel tubes support the piping system for the burst slug detection (BSD) system. A small section of the fuel channel extensions and the upper plate is seen at the top of Fig. 4.

The principal problem in designing a heterogeneous graphite core is that of dealing with dimensional changes which result from irradiation. It is well known that at temperatures below 300° C graphite grows when irradiated, but at the temperatures of interest in enriched power reactors, neutron irradiation causes graphite to shrink anisotropically[1, 2, 3, 4, 5, 6, 7, 8, 9].

The largest change takes place in the direction of extrusion of the graphite. The magnitude of the effect is not precisely known, but it appears that, during reactor life, it is likely to reach several per cent. in the region of maximum exposure to neutron irradiation. A graphite core and moderator could be made up of a number of pieces which would remain in close contact throughout life. A stack with EGCR dimensions might shrink by approximately 1½ in across its 15 ft 10 in diameter at the midplane and 3 in along its 19 ft 4 in axis. The resulting hour-glass shape would distort the fuel and control rod channels. Three problems then arise.

First, it is necessary to bind the stack around its girth with a restraining band to maintain its integrity as a unit; the band must exert its restraining force in both the cold and hot conditions. A restraint that can do this while it shrinks several per cent. and while unattended through 20 years of irradiation is not easy to design.

Second, as the graphite shrinks and the restraining band continually draws the individual pieces together, it becomes necessary to provide for slippage of the bottom surface of graphite across its steel supporting structure. At the same time, alignment with fuel channel entry tubes which pass through the support structure must be maintained.

Third, the alignment of control rod channels changes with respect to the control rod nozzles of the reactor vessel. This shift in alignment, combined with the distortion of the channel itself, may cause control rods to bind.

In view of these problems, it is desirable to design

the core as a group of autonomous graphite columns which can stand independently apart from each other as shrinkage occurs. Very recent advances in the technology of graphite manufacture have made it possible to obtain single pieces of graphite over 20 ft in length, and such pieces are employed in the EGCR as described above. It is found that the long graphite columns, even though properly positioned at top and bottom so as to maintain relationships between core channel and vessel access nozzles, will not behave respectably along the mid-section because of variations in shrinkage rate due in turn to variations in neutron flux across the individual block.

There are two superimposed effects of these flux gradients, which are illustrated in Fig. 5. It is seen that the actual flux shape may be regarded as the sum of a linear component plus a component of higher order variation. As the shrinkage is directly proportional to neutron irradiation, these curves may also be regarded as variations in strain. The linear component causes the block to bow, and if the bowing is prevented by external restraint the stress generated is proportional to the thickness of the block, so that the magnitude of the stress is subject to some control by the designer when he chooses the block thickness. In the EGCR the thickness of two lattice cells, 16 in, was selected as the best compromise between the reduced stress and reduced strength for handling which accompanies a small thickness-to-length ratio.

The other component of shrinkage variation is the complicated higher-order one due to the fine structure of the neutron flux within a lattice cell. The severity of this component cannot be significantly reduced even by slicing the graphite into strips smaller than the 8-in cell dimension. It is a problem that is inherent in the use of graphite as a moderator. The stress from this component is therefore not under the control of the designer of a graphite-moderated reactor.

The magnitude of the shrinkage rate is not accurately known. Extrapolation of the meagre short term data that is applicable to EGCR conditions indicates that stress induced by the higher order component of shrinkage variation may exceed the breaking strength of the graphite in three to five years. In operation it is expected to perform better than this because both its creep rate and its strength are believed to be enhanced by radiation. It may be that it will never break for these reasons.

In view of these uncertainties, Allis-Chalmers designed the core so that it would remain operable even if the graphite should break. It was calculated that the bowing of the columns because of the linear component of shrinkage variation might amount to several inches. Therefore it was decided to place restraining bands (Fig. 4) around the outside of the core. These bands only come into play when the graphite begins to bow. They limit the misalignment of any channel in the worst case to approximately 1 in. Should a block break and the separate sections then become misaligned for any reason, the misalignment is similarly limited. If this occurred, it would probably be necessary to broach or ream the channels in the affected block.

Primary Cooling System

The reactor vessel consists of an outer pressure shell of 2¼ in carbon steel plate, and an inner temperature barrier of 1 in stainless steel. The pressure shell is approximately 20 ft diameter by 46 ft high. The vessel is designed for 350 p.s.i.g. and 650° F.

The steam generators are to an original design of C. F. Braun and Co., of Alhambra, California, by whom they are also being built, in which the steam drum is spherical and welded into the upper end of a cylindrical vessel. The tubes pass through the bottom surface of the sphere, which acts as a tube-sheet, into the cylindrical helium-filled space. The helium flows into the side of the cylindrical section and across the tubes.

EGCR
(Experimental Gas Cooled Reactor)

Power	
Reactor thermal power	85.0 MW
Gross electric power	29.5 MW
Blower power	4.0 MW
Other auxiliary power	3.6 MW
Net electric power	21.9 MW
Coolant	
Coolant gas	Helium
Pressure, at blower exhaust	315 p.s.i.a.
Pressure, at blower suction	298 p.s.i.a.
Temperature, core outlet	1,050° F
Temperature, vessel outlet	1,043° F
Temperature, core inlet	510° F
Temperature, vessel inlet	500° F
Flow	427,000 lb/h
Core	
Moderator	Graphite monolithic columns
Specific power	7.55 MW(t)/t
Loaded fuel channels	216
Fuel assemblies per channel	6
Fuel elements per assembly	7
Lattice pitch	8 in sq.
Overall diameter	15 ft 10⅛ in
Overall height	19 ft 4 in
Equivalent active diameter	11 ft 10⅛ in
Equivalent active height	14 ft 6 in
Control rods	21
Fuel	
Composition	UO_2 sintered pellets
Enrichment, start up	2.24%
Enrichment, feed	2.55%
Cladding	304 SS
Cladding thickness	0.020 in
Diameter, elements	0.75 in OD
Diameter, sleeves	5.0 in OD
Diameter, sleeves	3.0 in ID
Maximum unperturbed heat flux	105,000 B.t.u./ft²h
Average heat flux	68,500 B.t.u./ft²h
Pressure Vessel	
Vessel material	Carbon steel SA 212B
Design pressure	350 p.s.i.g.
Design temperature	650° F
Inside diameter	20 ft
Inside height	45 ft 8 in
Thickness, wall	2¾ in
Thickness, head	4 in
Temperature barrier material	304 SS
Temperature barrier thickness	1 in
Steam Plant	
Type	Single pressure, non re-heat
Throttle pressure	1,265 p.s.i.a.
Throttle temperature	900° F
Turbine heat rate	10,180
Feedwater heaters	3
Condenser pressure	1.5 in Hg abs.

Steam Plant

The 25,000-kW steam turbine serves primarily as a heat sink while the reactor is operated at constant power as a test facility. The steam conditions of 1,250 p.s.i.g. and 900° F. were not determined by economic optimization but by minimization of out-of-pocket expense. It is planned that scheduled runs as a load-following power plant will be made to demonstrate the suitability of the reactor for this type of service.

The Detection of Failed Elements

The system for the detection of failed fuel elements (BSD) is based upon the British systems, which by now are well developed. Such a system employing intermittent sampling was included in the EGCR in the belief that it is necessary although not sufficient. Its principal limitations are two:

(1) The intermittent sampling may permit severe failures which occur between samples to raise the background count of the coolant so that the activity from the affected channel is not detected when its sample is later taken.

(2) The dependence upon direct fission product recoils does not permit detection of failures which do not expose fuel directly to the gas stream.

One of the important purposes which the EGCR will serve is operational experimentation to find practical means for overcoming these deficiencies. In addition, it would of course be desirable to increase the sensitivity of the system, which is expected to be about one square millimetre of exposed fuel surface.

The BSD piping system is to be combined with a pneumatic temperature monitoring (PTM) system by the provision of two sonic nozzles and appropriate instrumentation. One sonic nozzle is located near the point at which the gas sample is taken from the fuel channel extension, while the other is located outside the biological shield. The mass flow through a sonic nozzle can easily be

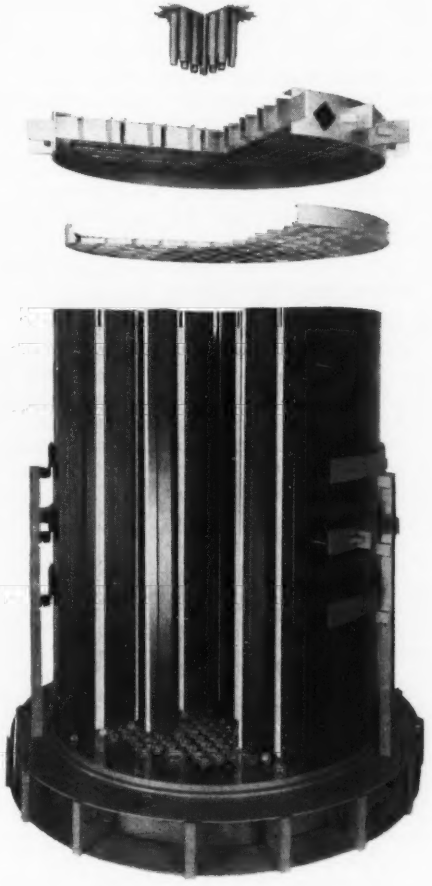


Fig. 4.—Model of the core.

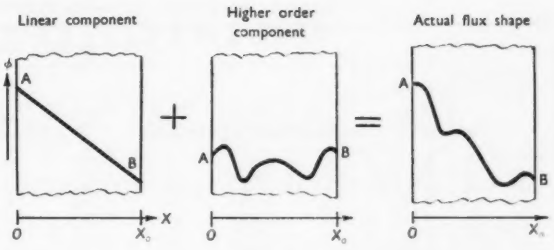


Fig. 5.—Variation in neutron flux and radiation-induced strain across a graphite block. ϕ neutron flux, or radiation-induced strain; X distance across block; X_0 width of block.

expressed in terms of the pressure ratio across it, the specific heat ratio of the gas, and the temperature on one side. Knowing the conditions on both sides of the nozzle which is outside the vessel, and the pressure inside, it is possible to solve for the gas temperature on the inlet side of the inside nozzle. This is done by equating the mass flow through the two nozzles.

The complete PTM system will not be installed during construction, but only those parts of the system inside the biological shield. The remainder of the system can be installed and placed in operation at a later date.

Charging and servicing machinery

In view of the unknowns involved in building a new type of reactor on a short schedule, it was decided that access should be possible to both ends of the core during operation. The charge machine is located beneath the reactor as the fuel can be changed faster from the cold end and with the aid of gravity.

A service machine, provided above the reactor, can also handle fuel but at a reduced rate. However, its primary function is the installation and removal of control rods and instrumented fuel assemblies as well as dropped or jammed components. The machine will also insert and remove experiments in the loops.

Because the machine is to handle active experiments, it will require decontamination at times. Since there are no known lubricants which can withstand the decontamination process, it was decided to design for operation with a minimum of lubrication. For this reason, the bearings are derated by a factor of 15 from manufacturer's ratings and their Hertz stresses are kept below 100,000 p.s.i. wherever possible. Contact stresses in the gears are even less. It is anticipated that the machine will have some useful life, even if operated in the completely unlubricated condition.

On-stream replacement of failed fuel elements

During the design of the EGCR fuel element, a statistical analysis was made of the distribution of fuel temperatures[9, 10]. The variation of fuel surface temperatures during normal operation is shown in Table 1. It is seen that 29 fuel elements of the 9,072 in the reactor are expected to experience local temperatures above 1,600° F. Combination of each of these probable temperatures with the creep strength of the cladding at that temperature permits calculation of the probable fuel failure rate. It is assumed that 50% of the fission gas is released from the fuel and that failure occurs when the cladding has expanded through creep to decrease channel flow area by 1%. A probable failure rate is then obtained of one element every twelve days.

It is important to recognize that both the rate of gas

TABLE 1. MAXIMUM PROBABLE CLADDING TEMPERATURES FOR FUEL ELEMENTS

Temperature range, °F	FUEL ASSEMBLY NUMBER (Top Assembly is No. 1)												Total elements
	1		2		3		4		5		6		
	Probability	No. of fuel elements	Probability	No. of fuel elements	Probability	No. of fuel elements	Probability	No. of fuel elements	Probability	No. of fuel elements	Probability	No. of fuel elements	
Below 1,300	0.0135	20	0.0117	18	0.0900	136	0.9992	1,511	1.0000	1,512	1.0000	1,512	4,709
1,300—1,350	0.0899	136	0.0468	71	0.1842	279	0.0008	1	0.0000	0	0.0000	0	487
1,350—1,400	0.2445	370	0.1424	215	0.2656	402	0.0000	0					987
1,400—1,450	0.3845	582	0.2582	390	0.2483	375							1,347
1,450—1,500	0.2082	315	0.2782	421	0.1451	219							955
1,500—1,550	0.0532	80	0.1789	270	0.0529	80							430
1,550—1,600	0.0059	9	0.0668	101	0.0120	18							128
1,600—1,650	0.0003	0	0.0149	23	0.0017	3							26
1,650—1,700	0.0000	0	0.0019	3	0.0002	0							3
1,700—1,750			0.0002	0	0.0000	0							
1,750—1,800			0.0000	0									

release and the failure criterion which are assumed for purposes of analysis are arbitrary. Tests are in progress to provide better bases; however, analysis of this type can give useful indications to the designer as to whether his equipment capabilities are of the right order of magnitude.

There are sound grounds for arguing that the calculated failure rate is of the proper order of magnitude for a commercial power plant. If the requirement were placed upon the fuel that the probable failure rate should be once in five years, say, so that fuel would not have to be replaced during operation, the fuel temperatures would have to be reduced approximately 125° F, which would reduce the gas outlet temperature approximately 75° F. If this temperature loss resulted in a 5% increase in plant heat rate, the capitalized value of this decrease in efficiency in a $50 million power plant would exceed $2,500,000 which is the amount available for investment in equipment to detect and remove failed fuel elements during operation.

On the other hand, it is easily seen from the table of fuel temperatures that an increase in fuel temperature of 50° F would sharply increase the failure rate. This increase might require provision of another machine, while the improvement in plant efficiency would only be approximately 2%. Thus, the economy of such increased demand on the fuel is immediately open to question. One of the important functions of the EGCR will be the provision of actual operating experience upon which such decisions can be made for future reactors.

While the above argument establishes the economic basis for the capability to detect and remove failed elements, there is an additional general basis, and in the EGCR this capability is only a part of an over-all emphasis on flexibility. In the beginning of the EGCR design effort, Allis-Chalmers adopted the basic criterion that the operator should have the wherewithal to detect, locate and remove failed fuel elements, and that the design should include the maximum operational flexibility. In this way latitude is allowed for operator ingenuity to overcome problems and difficulties which the designer did not have the prescience to foresee. This was the basic reason for the provision of graphite sleeves to support the fuel, the PTM system to monitor channel temperatures, the BSD system to detect and locate fuel failures, the on-stream charging feature, and the provision of machines with access to both ends of the reactor during operation. In short, the designer's calculations may be in error and all expectations may not achieve complete realization, but the operator will have at his disposal the maximum capability to deal with unexpected problems.

Fig. 6.—The site of the EGCR at Oak Ridge in August, 1960.

On-stream replacement for exhausted fuel

While the above reasoning justifies, for a commercial power plant, the equipment to detect, locate and replace failed fuel, nothing has been said to show that the continuous charging of fresh fuel is also necessary. One is likely to assume that because the first capability is included the second follows from ease of design and economy of construction. This is the kind of assumption that is easy to make during preliminary design, but may be difficult to live with as one faces the realities of detailing for construction. An economic analysis performed by Allis-Chalmers showed that the capability to replace exhausted fuel while operating would definitely be justified in a full-scale power plant. This capability was therefore incorporated in the EGCR design.

Schedule for Construction

The status of the containment shell and the buildings in August, 1960, is shown in Fig. 6. The bottom portion of the reactor vessel will be installed in the spring of 1961, followed by the installation of the graphite core. The top head of the vessel will then be welded on. Construction is to be complete in the summer of 1962. The reactor is expected to be critical late in 1962, with full-power operation expected by the end of that year or early in 1963.

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Nuclear Ship Propulsion

Taormina, Sicily
November 14—18

Recent IAEA Symposium Pays Special Attention to Nuclear Safety

THE first session of the Conference was devoted to economics and national activities in nuclear ship propulsion. Both the papers presented and the subsequent discussion showed that it is now generally accepted that nuclear propulsion has no immediate promise of being economic for merchant ship propulsion, but the belief is widespread that this new power source will ultimately replace the use of oil for all but the smallest power installations. It is believed that nuclear fuels will eventually produce power at a price considerably less than that for oil and that the capital cost of nuclear installations will be progressively reduced. Furthermore, the view was more than once expressed that the quickest way to make progress in this field was to build a ship and get operating experience with a type of reactor already developed on land, irrespective of whether this type of reactor will eventually find favour for the marine application; the thought being that operating experience will lead to a reduction in the complexity and the safeguards adopted.

A paper by L. W. Boxer of OEEC presented an ENEA survey of nuclear ship propulsion activities in Western Europe. This paper suggests that experience could be accumulated by sea-going European nations by pooling of present resources to share in the cost of construction of a European nuclear ship as a joint undertaking of the ENEA. In this context a valuable step forward is already envisaged by Euratom in their projected participation in the national ship propulsion research and development programmes of France, Germany and Italy (with the possible inclusion of the Netherlands). A paper presented by M. Yamagata of Japan surveyed Japanese activities in nuclear propulsion. Although the 1960 budget did not include the expenses for the first experimental nuclear ship a considerable number of researches are being made by governmental and private organizations—many of which are subsidized by the Government—into various problems connected with nuclear propulsion. As in several other countries the main interest at present appears to lie in the boiling water reactor.

The next two sessions were devoted to international problems and general aspects of safety for nuclear ships. A paper by H. N. E. Whiteside reviewed the new regulations and recommendations applicable to nuclear-powered merchant ships agreed at the 1960 SOLAS Convention in London and a paper by Capt. C. P. Murphy of the United States Coastguard discussed port entry, control and safety assessment provisions of this Convention as applied to the N.S. *Savannah*. The N.S. *Savannah* Safety Assessment consists of four volumes. The first volume entitled "Engineering and Construction" and the fourth volume entitled "Analysis of Hypothetical Accidents" have already been issued, but the second volume "Operation" and the third volume "Radiological Health" will not be issued until sea trials and some experimental operation have provided data that will be useful to foreign officials in the evaluation of the safety of the ship. The safety assessment is not designed for general public use nor for technical press interpretation and its distribution has therefore been limited to Government exchange.

Papers were also presented on the classification standards for nuclear-powered ships of Lloyd's Register of Shipping and also Bureau Veritas. An excellent survey of these standards was presented by N. Krarup of Danatom entitled "Evaluation of Rules and Regulations for Nuclear Ships." Important differences in requirements lie in the maximum shock accelerations for which the reactor should be designed and in particular the requirement of Lloyd's Register for design on the basis of 3g in any direction applied through the supports and connections to the hull structure is exceptional. Considerable differences also exist in the design requirements for the containment vessel. The *Savannah* containment vessel is designed to withstand an internal pressure of 173 p.s.i., the highest calculated pressure following the maximum credible accident, and the vessel has been hydraulically tested to this pressure at the bottom. It would appear that in this test the containment vessel dome was in fact tested to 173 p.s.i. less the static head, that is about 150 p.s.i. Lloyd's Register require a hydraulic test to 125% of the maximum pressure.

For the water reactors the maximum credible accident is a major break in the primary circuit leading to a partial melt-out of the core and the main concern is the leakage of fission products from the containment vessel. The *Savannah* is provided with a ventilation system to keep the pressure in the reactor compartment below normal, the air being passed through filters and discharged at the top of the mast.

Fracture of the containment vessel is not envisaged in any circumstances near land and the philosophy adopted for the *Savannah* was indicated in the following paragraph, extracted from a paper entitled "Safety Consideration in Nuclear Ships" by C. K. Beck and E. G. Case of the Division of Licensing and Regulation of the U.S. AEC.

"In estimating the probability of a dangerous collision involving the *Savannah*, it has been found that practically all collisions involving passenger ships take place within 'approach' areas. This area is defined as that extending from the principal sea mark at a harbour entrance to the 'point of departure from a port,' 100 miles from the nearest land. The area is characterized by the merging of sea lanes at harbour approaches, heavy traffic, and relatively high ship speeds. Accordingly, it can be seen that this area of highest collision probability begins at a point usually several miles distant from a harbour, and that the exposure risk of the general population from a collision in this area is reasonably small. In addition, it can also be shown that because of containment and anti-collision features surrounding the *Savannah* reactor, the probability of damaging collision inside a harbour is also acceptably small, as ship speeds are so low as to preclude penetration of the reactor compartment within these areas."

Ten papers were presented on nuclear ship projects from the angle of safety and of these five were concerned with pressurized and boiling water reactors, two with organic liquid reactors, two with advanced gas cooled reactors and one with a steam cooled marine reactor. A number of investigations are in progress on the stability of boiling water reactors in a seaway but the matter still appears to be unresolved. In this connection it was interesting to hear that the Norwegian Institute for Atomic Energy Research have for some time been working on a simulator project better known as the "Rock 'n Roll Project." In the Rock 'n Roll Project a floating sphere of 24 metres diameter has been proposed as a combined reactor system containment and support vessel in order that a controlled pitching and rolling motion may be imposed upon the plant with the minimum of mechanical effort. The reactor centre line would be offset at a horizontal distance of about 9 metres from the sphere centre and the sphere could be trimmed by seawater ballast and oscillated at its natural frequency about any horizontal axis through the sphere centre.

The view was put forward in one of the papers on the organic liquid moderated reactor that a core melt-down was incredible. This contention arises from the fact that the bulk

coolant temperature is less than the boiling point at atmospheric pressure and it is difficult to foresee circumstances in which the coolant is lost from the core. A high degree of inherent safety can be exhibited by gas cooled reactors and in a paper entitled "Safety Aspects of the Marine Advanced Gas Cooled Reactor" by J. O'Neill and M. C. Hartnell Bevis of the General Electric Company the claim was made that even with a very rapid reduction in coolant flow the fuel cans would not melt whether the safety rods were operating or not.

A session was devoted to ship reactor problems and under this heading papers were presented on the shielding of marine reactor installations and also on the manœuvrability of nuclear powered ships. A Japanese paper gave the results of a study of vapour suppression for the container of nuclear powered ships and the similar proposals were made in the paper entitled "Some Safety Problems in the Ansaldo Fiat Nuclear Ship Study" by L. Chinaglia and P. Rucci. Basically the system provides a heat sink to condense the steam flashing in case of the failure of the primary circuit of a pressurized water or boiling water reactor and enables the maximum pressure to be reduced to a value where normal ship construction methods can be used for the containment. In the event of a loop break the steam flows through ducts and is condensed by cold water in cofferdams situated inside the containment.

Two papers were presented on accelerations due to ship motions and measurements made on board the M.S. *Hodakasan-Maru* showed a maximum acceleration at the bow by slamming of approximately 3*g* while the maximum acceleration at amidships was estimated at about 0.9*g*.

An interesting paper was presented on the maintenance planning for the N.S. *Savannah* and it was disclosed that although the first core is likely to have a lifetime in excess of three years a decision will be made in 18 months' time whether or not to refuel the *Savannah* then with the second core of more advanced design. Routines are also being worked out for the renewal if necessary of such items of equipment as the primary pumps. In the discussion on this paper a suggestion was made that ingress of seawater to the secondary circuit could possibly lead to chloride attack on the stainless steel tubes of the heat exchanger. The answer was put forward that single tube failures would be plugged but that the retubing of a heat exchanger would be a major operation and the routine for such had not yet been worked out.

The last session of the conference was devoted to the safety aspects of nuclear ship operation. A paper by I. C. Roberts of the International Atomic Energy Agency surveyed waste disposal problems in nuclear ship propulsion. The subject of radioactive waste disposal into the sea is continuing to receive very active interest by the IAEA. The *ad hoc* panel of experts which studied the radioactive waste disposal into the sea, the Brynielsson panel, made certain recommendations which have an international character and in January, 1961, a panel of legal experts will meet in Vienna to study the legal factors involved. The IAEA are also studying the factors and procedures required for adequate harbour evaluation and hopes to be able to provide assistance, in the form of experienced personnel, to countries wishing to make harbour evaluations. In addition, a group of experts have been invited to meet in Vienna in April, 1961, to study means by which oceanographic radiation monitoring techniques can be standardized. It was pointed out in discussion that the convention on oil pollution had not completely overcome this problem and therefore the radioactive waste disposal at sea should be treated with some degree of caution. Mr. Morgan, of the Ministry of Agriculture and Fisheries, stressed the need for detailed research in this field to trace each isotope dispersed into the sea back to man.

A very thoughtful paper on the safety considerations arising from the operation of nuclear merchant ships in ports was given by G. D. Bell of the U.K.AEA and two further papers were also concerned with this subject in somewhat less detail. It is apparent that a very high degree of retention of the fission products present in a propulsion reactor is required to avoid excessive exposure of the general population and, in addition, it is desirable that any release takes place over a period of hours rather than minutes so that adequate measures can be taken in good time to remove the working population in the immediate vicinity of the ship. The necessary restriction on the escape of fission products can be obtained by a variety of methods used singly or more generally in combination. In some systems it may prove feasible to provide designs in which the fuel melt-out can be prevented in the event of failure of the primary circuit. Where a conventional pressure vessel type of containment is employed, additional measures, such as recirculating clean-up systems, the use of the ship's hull as an additional hold-up space, and controlled ventilation or depressurization may be employed. At the present time no standards have been laid down for the permissible discharge of radioactivity either from the point of view of the maximum dose received by a member of the civil population in any such emergency or the total dose received by the population in the vicinity, the latter being important from the point of view of genetic effects. Many national and international bodies are engaged on the study of acceptable emergency levels of exposure for environmental populations and it may be expected that in due course international standards will be recommended. The further work involved in translating these exposures in terms of air concentration or contamination level should eventually lead to values which can be used as a basis for design to internationally accepted standards. While insisting that appropriate safety standards are enforced it may be considered prudent to select those ports, and those berths within a given port, which offer the greatest factors of safety against a release of activity and in which the problems of protection of the general public are not unduly severe. There remains the remote danger of high speed collision or the possibility of stranding in the port approaches and particular ports must be studied with this aspect in mind, having regard to the structural protection and strengthening which the designers may have provided. From the discussion of this subject of port assessment it was apparent that conditions varied widely in different ports and it seems likely that nuclear powered ships will be restricted to a limited number of favourable ports until more experience has been obtained.

N.S. Savannah in course of being fitted out at the Camden, New Jersey, yard of New York Shipbuilding Corporation.

NEW BOOKS

Radioactive Wastes; Their Treatment and Disposal. Edited by John C. Collins, B.Sc., M.S.E., A.M.I.C.E. (239 pp., 46 illus. E. and F. N. Spon, Ltd. 55s.)

Authorship, like research, is becoming more and more a matter of teamwork rather than individual effort, and this book is another example of the "symposium" treatment. While opinions of the advantages of this system may vary with some subjects, there can be little doubt of its value in the case of a comparatively new field such as wastes engineering, which is developing so rapidly that it bids fair to become a separate branch of engineering, and has so many different aspects that no single person could hope to be an authority on all of them.

Since interest in this subject is not confined to those actually engaged in the disposal of radioactive wastes, but ranges over a field extending from municipal engineers to marine biologists, and from water supply executives to factory inspectors, it is satisfying to observe that the contributors represent a wide cross-section of interests. The nature of radioactivity (Chap. I) is covered by J. A. Barnard of the Department of Chemical Engineering of University College. Chap. II, on the sources of radioactive wastes, is dealt with by W. T. L. Neal, of the Ministry of Housing and Local Government, while R. F. Farr, of Queen Elizabeth Hospital, Birmingham, reviews radiation hazards, and F. W. Cornish, of AWRE, Aldermaston, contributes a chapter on the measurement of radioactivity. A chapter on the law of radioactive wastes in Britain, by R. H. Burns and W. J. Megaw of Harwell, concludes what is really the first half of the book; the basic principles necessary to a thorough understanding of the subject.

The second half of the book is devoted to the practical problems, methods and techniques of disposal work. The treatment of radioactive liquid effluents is dealt with by R. H. Burns, of Harwell, who is also the author of the chapter on the disposal of solid wastes. The actual disposal of liquid effluents is dealt with by J. C. Collins, of Manchester University. The biological concentration of radioactivity, and its application to the treatment of liquid effluents is the province of G. E. Eden, of the Water Pollution Research Laboratory, while W. J. Megaw, of Harwell, contributes the final chapter on gaseous discharge to the atmosphere. In addition to the index, there is a glossary of terms which will be particularly useful in a book intended for a mixed readership. The thorough coverage of the subject is attested by nearly 200 references to existing and unpublished work in this field. N.E.T.S.

Materials for Nuclear Engineers. Edited by A. B. McIntosh, Ph.D., F.R.I.C., A.R.C.S.T., F.I.M., and T. J. Heal, M.Sc., F.Inst.P., F.I.M. (384 pp. Temple Press Limited. 75s.)

The importance of materials in nuclear engineering scarcely requires stressing. This book has the advantage of being a "composite" work in which each of the sections is compiled by an expert in the particular field. It collects in one convenient volume the information pertaining to reactor fuels, moderators and cans which appeared in the popular series in *Nuclear Engineering* between 1957 and 1959, and has been edited and brought up to date. Contents include an introductory section, "Materials Information—Its use in Nuclear Engineering," by A. B. McIntosh, and the following individual material studies: Uranium, by R. W. Nichols, B.Met., A.I.M.; Plutonium, by K. Q. Bagley, B.Sc., Ph.D.; Thorium, by I. P. Bell, M.Sc., A.Inst.P.; Ceramic Fuels, by R. W. Nichols, B.Met., A.I.M.; Graphite, by W. G. O'Driscoll, M.Sc., and J. C. Bell, B.Sc., A.Inst.P.; Magnesium, by T. J. Heal, M.Sc., F.Inst.P., F.I.M.; The Metallurgy of Beryllium, by L. R. Williams, B.Sc., F.I.M., and P. B. Eyre, B.Sc., A.I.M.; Zirconium, by C. Tyzack, B.Sc., Ph.D. Each section is complete with a wide selection of references, and the book is well indexed.

The authors are all on the staff of the United Kingdom Atomic Energy Authority and at the time of writing the original draft were closely connected with the activities of the Culcheth laboratories, the prime function of which was to undertake research and development within the Industrial Group to permit the selection and specification of materials for reactor projects and the chemical and processing plants. As a result they have been in an especially advantageous position for collecting and assessing information pertaining to the properties of the important nuclear engineering materials.

Following the publication of the original articles in *Nuclear Engineering* a great deal of new information came to light, in particular at the 1958 Geneva conference. This has been carefully analysed and the necessary modifications made to the original texts —in some instances involving, in effect, a complete rewriting. As a result the book represents the most up-to-date digest of the properties of the materials considered that is available.

Somewhat naturally the book leans towards the gas cooled graphite moderated technology but not exclusively and the authors have been careful to include as much overseas information as possible where this is relevant. The section dealing with ceramic fuels is of necessity somewhat brief and in particular the section on uranium oxide can do no more than pierce the surface of the large body of data that has been accumulated, especially in the U.S.

It is important to note that although the book has been compiled by metallurgists it has been written for the engineer and is designed to provide him with a reference work prepared in a language with which he is familiar and in a form that is directly relatable to engineering problems. In bridging the gap between the two disciplines its success is noteworthy.

Atomic Radiation and Polymers. By A. Charlesby. (556 pp., 212 illus. Pergamon Press, Ltd. 110s.)

Professor Charlesby was among the first of the world's scientists to investigate in depth the influence of radiation on polymers. As a result of investigational work carried out or instigated by him in Government institutes and Tube Investments, Ltd., we now have at our disposal an immense fund of information on the effects of submitting various polymers to various types of radiation.

As a subject, it can be considered in two ways; first as a route to novel types of polymers as a result of changing their structure by exposing them to radioactive sources of one type or another. The second is to regard the radiation of polymers as a means of investigating the structure of polymers and thus exploring their structure in a manner which would otherwise be very difficult.

For those to whom the first concept is of importance this last work by Professor Charlesby may be regarded as of prime importance. It reports exhaustively on the present state of knowledge of the results of radiation treatment on such polymers as polythene, polystyrene, silicones, p.v.c., polypropylene, polymethylmethacrylate, p.t.f.e., cellulose materials and various types of rubber. To this well-prepared text is added a very full bibliography which will make the task of the investigator very much more simple.

To those who regard radiation as an investigational tool this book may be regarded as a useful, although not so exhaustive work. Professor Charlesby has managed to introduce a considerable quantity of polymer chemistry and physics as a background to his discussions on the changes induced in organic molecules by radiation. The literature references that he has cited, however, more than compensate for the necessary condensing that has been called for in the treatment of such a large subject.

In the purely specialized sense, the chapters he has included on radiation protection and the use of polymers in nuclear reactors is a valuable contribution to the literature. Polymers are high cost materials and their extensive use for protection purposes would be limited, presumably, on this score alone. From the scientific point of view, however, polymers exhibit such novel characteristics in this area as to make the investigation of their properties all-essential. The fact that Professor Charlesby has assembled so much data on this subject is therefore a matter of great satisfaction.

Scientists in the nuclear and high polymer field will find Professor Charlesby's book a most valuable addition to their libraries and a landmark in the development of this very specialized field of investigation.

C.W.W.

Books Received

Practical Nucleonics. By F. J. Pearson, B.Sc., Ph.D., A.Inst.P., and R. R. Osborne, B.Sc., Dip.Ed., A.Inst.P. 208 pp. Spon, Ltd. 37s. 6d.

Progress in Nuclear Energy, Series IV: Technology, Engineering and Safety, Vol. 3. Edited by C. M. Nicholls. 456 pp. Pergamon Press, Ltd. 105s.

Resonance Absorption in Nuclear Reactors. By Lawrence Dresner. 131 pp. Pergamon Press, Ltd. 40s.

Translation from Russian for Scientists. By C. R. Buxton and H. S. Jackson. 299 pp. Blackie and Son, Ltd. 30s.

Theory of Thermal Stresses. By B. A. Boley and J. H. Weiner. 586 pp. John Wiley and Sons, Ltd. 124s.

Progress in Nuclear Physics. Vol. 8. Edited by O. R. Frisch. O.B.E., F.R.S. 304 pp. Pergamon Press, Ltd. 90s.

Nuclear Forces and the Few Nucleon Problem. Vols. I and II. T. C. Griffith and E. A. Power. 720 pp. Pergamon Press, Ltd. 200s.

Thermal Reactor Theory. By A. J. Galanin. Translated from the Russian by J. B. Sykes. (412 pp. Pergamon Press. 100s.)

TRANSLATIONS

A Résumé of Principal Articles

Embases minces pour échangeurs de chaleur haute pression. (8)

Une analyse théorique, dans le numéro précédent, a montré que l'utilisation d'embases très minces (16 s.w.g.) est possible pour des échangeurs de chaleur travaillant à 70,3 kg/cm² bien que les contraintes soient extrêmement sensibles aux petites variations de température.

Les expériences effectuées, pour vérifier cette théorie, sont décrites dans le présent numéro. Les valeurs des contraintes dans les embases minces sont données pour quelques expériences préliminaires. Celles-cis montren que les embases minces sont plus résistantes que la théorie ne le prédit.

Trawsfynydd. (12)

Trawsfynydd, la cinquième centrale nucléaire civile, a plusieurs caractéristiques intéressantes. L'une des principales est peut-être la grande puissance des générateurs (145 MW), ainsi que l'utilisation de circulateurs à vitesse constante, le contrôle du débit du gaz réfrigérant étant effectué à l'aide d'un circuit auxiliaire. Il est intéressant de mentionner qu'il s'agit de la première centrale nucléaire britannique non située au bord de la mer qui ne soit pas pourvue de tours de réfrigération: le refroidissement nécessaire est obtenu grâce à un lac artificiel.

Protection contre les radiations à Trawsfynydd. (16)

Tout en respectant les normes, il est possible d'arriver à une solution économique par une bonne disposition et en combinant de façon économique les matériaux de protection. Les problèmes se rapportant à la protection de la machine de chargement des èlements de combustible ainsi que les principales caractéristiques du blindage biologique sont également discutés.

Les causes de fonctionnement défectueux à Trawsfynydd. (18)

Lorsque l'on étudie un réacteur de puissance, toutes les causes de fonctionnement défectueux probables doivent être analysées d'emblée, de sorte qu'un compromis harmonieux entre l'économie, les prestations et la sûreté de fonctionnement puisse être réalisé.

L'arrangement du graphite des réacteurs de Trawsfynydd. (21)

Le but recherché est d'assurer l'intégrité des canaux en toutes circonstances, aussi bien lorsque les conditions sont normales qu'en cas d'accident. En conséquence, il sera toujours possible d'actionner les barres de contrôle, les éléments de combustibles pourront être insérés ou retirés, enfin les fuites de réfrigérant dans les canaux, seront réduites au minimum. Les déformations des canaux ne seront pas importantes, pourvu que les conditions réalisées correspondent aux hypothèses retenues lors du projet.

Un coeur résistant aux tremblements de terre pour Tokai-Mura. (25)

L'étude d'un coeur capable de résister aux contraintes additionnelles dues à un tremblement de terre a été le problème le plus difficile qu'il a allu résoudre pour le réacteur de Tokai-Mura.

Dünne Rohrplatten für Hochdruck-Wärmeaustauscher. (8)

Im Teil I wurde auf Grund einer theoretischen Analysierung gezeigt, dass die Verwendung sehr dünner Rohrplatten (16 s.w.g.) für Wärmeaustauscher, die mit 1000 p.s.i. (70,3 kg/cm²) arbeiten, möglich sein dürfte, obwohl die Beanspruchungen ausserordentlich empfindlich schon gegen kleine Temperaturschwankungen zu sein schienen. Teil II beschreibt einige der experimentellen Arbeiten, die unternommen wurden, um die Theorie zu prüfen, und gibt vorläufige Resultate für die Beanspruchungen in dünnen Platten aus Metall. Die Experimente zeigen, dass dünne Rohrplatten stärker sind, als die Theorie voraussagte.

Trawsfynydd. (12)

Trawsfynydd ist das Fünfte der auf kommerzieller Grundlage arbeitenden Kraftwerke und hat viele interessante Charakteristiken, von denen möglicherweise zwei besonders interessieren, nämlich die bedeutende Grösse der Generatoren (145 MW) und die Verwendung von Gebläsen mit nur einer Geschwindigkeit, wobei zur Kontrolle der Kühlung Nebenkanäle für das Gas vorgesehen sind. Ein weitere interessante Tatsache ist die, dass es das erste Kernkraftwerk in Grossbritannien ist, dass nicht an der Küste liegt und keine Kühltürme hat, sondern von einem künstlichen See her seine Kühlung empfängt.

Abschirmung gegen Strahlung in Trawsfynydd. (16)

Unter Beachtung aller Sicherheitsvorschriften der Strahlungkunde können niedrige erste Anlagekosten durch gute Planung und die richtige Wahl wirtschaftlicher Abschirmmaterialien erzielt werden. Der Aufsatz enthält auch eine Diskussion der Erfordernisse für die Abschirmung der Maschinerien für die Versorgung mit Brennstoff und der Konstruktion der haupsächlichen biologischen Abschirmung.

Bedingungen für Fehler in Trawsfynydd. (18)

Beim Entwurf eines Kraftwerk-Reaktors müssen alle möglichen Bedingungen für Fehler in einem frühen Stadium geprüft werden, sodass das richtige Gleichgewicht zwischen Wirtschaftlichkeit, Leistung und Sicherheit gefunden werden kann.

Die Konstruktion des Graphit-Kerns. (21)

Der Gesichtspunkt, unter dem der Kernentwurf gemacht wurde, war der, dass die Kanäle unter allen Betriebsbedingungen einwandfrei bleiben sollten, auch wenn irgendwelche möglichen Fehler vorkommen sollten, sodass die Kontrollbohrungen betriebsfähig blieben, Brennstoff-Elemente eingeführt und entfernt werden könnten, und irgendwelche Kühlmittelverluste aus den Kanälen für die Brennstoff-Elemente durch Lecken innerhalb von Mindestgrenzen gehalten würde. Verwerfungen der Kanäle selbst werden nicht für wichtig erachtet, vorausgesetzt, dass die Konstruktion den oben erwähnten Bedingungen entspricht.

Der Reaktor Kern für Tokai-Mura, der gegen Erdbeben sicher sein muss. (25)

Eines der Hauptprobleme, die mit dem Reaktor Tokai-Mura verbunden sind, bezieht sich auf den Entwurf eines Kerns, der im Stande ist, die zusätzliche Belastung auszuhalten, die bei einem Erdbeben entstehen.

Planchas de Poco Espesor para Cambiadores de Calor de A.P. (8)

En Parte I un análisis teorético demostró que el uso de planchas tubulares de muy poco espesor (16 s.w.g.) en cambiadores de calor funcionando a 1.000 libras por pulgada cuadrada (70,3 Kg/cm²) podría ser posible, aunque los esfuerzos parecían ser extremadamente sensibles a los pequeños cambios de temperatura. La Parte II describe algunas labores experimentales llevadas a cabo para comprobar la teoría, juntamente con los resultados preliminares sobre los efectos del tensionado excesivo de planchas metálicas de poco espesor. Los experimentos indican que las planchas tubulares de poco espesor son más resistentes de lo que la teoría predice.

Trawsfynydd. (12)

Trawsfynydd, la quinta de las centrales de fuerza nucleares comerciales, posee muchas interesantes características, de las cuales posiblemente las dos más interesantes sean el gran tamaño de los generadores (145 MW) y el empleo de sopladores de velocidad única con el desviado de gas para control del enfriante. Otro hecho interesante es que es la primera central nuclear en la Gran Bretaña que no ha sido situada en la costa y que no tiene torres de enfriamiento, habiéndose obtenido este requisito de un lago artificial.

Blindaje contra la Radiación en Trawsfynydd. (16)

Con sujección a las exigencias radiológicas de seguridad, se puede conseguir un costo capital general reducido mediante buena disposición y la selección de combinaciones económicas de materiales de blindaje. También se discuten las exigencias de blindaje particulares de la máquina de alimentación de combustible y el diseño de la pantalla biológica principal.

Condiciones de Falla en Trawsfynydd. (18)

En diseñar un reactor de energía se deben examinar todas las posibles condiciones de falla en una etapa temprana, de modo que se pueda obtener un equilibrio apropiado entre economía, ' performance ' y seguridad.

Diseño Ingenieril del Núcleo de Grafito. (21)

La finalidad del diseño del núcleo en Trawsfynydd es la de mantener la integridad de los canales bajo todas las condiciones de funcionamiento e investigar cualesquier fallas posibles, de modo que los agujeros de control queden operacionales, que los elementos de combustible puedan ser colocados o retirados, y que la fuga del enfriante de los canales de los elementos de combustible sea la mínima posible. Las distorsiones de los canales, como tales, no se consideran de importancia, siempre que se cumplan con las condiciones estipuladas.

Un Núcleo de Reactor Resistente a Terremotos para Tokai-Mura. (25)

El mayor problema asociado con el reactor de Tokai-Mura ha sido el diseño de un núcleo capaz de resistir las cargas adicionales que ocurren durante un terremoto.

Technical Papers and Publications

IRON AND STEEL INSTITUTE

Symposium on Steel for Reactor Pressure Circuits.

Church House, London, Nov. 30-Dec. 2.

THE 13th Hatfield Memorial Lecture was given this year on the evening of Nov. 29 by Mr. L. Rotherham of the Central Electricity Generating Board, who chose as his subject "The Contribution of Metallurgy to Electric Power Generation." On the following day, the Symposium proper, which concentrated largely on gas-cooled reactors, was opened by Sir Leonard Owen with a talk entitled, "Steel and Nuclear Power." In this he pointed out that the limitation to the output of the Calder reactors was governed not by the canning material so much as by the steel available for the construction of the pressure vessels. The steel had to be suitable for site welding in very large sizes and as a result have good notch ductility properties, but if the full potential of the magnesium fuel cans were to be exploited the steel required a good creep resistance in the 400°C region. In the event, a grain controlled mild steel was chosen and the operating pressure had to be limited to 100 lb and the maximum temperature to 345°C.

As good notch ductility is vital and is adversely affected by nuclear radiation, the U.K. AEA has instituted a broad programme of work to determine the specifications that can be written into a steel and has concluded that the Robertson thick plate constant temperature test is the one best suited to determine the temperatures above which cracks will not propagate. This can not be applied as a routine test however and correlations with the Charpy test have indicated that this method can be used for normal test work.

Very thick plates introduce difficulties right from the steel making to final operation, but Sir Leonard emphasized that if better steels had been available good use could have been made of them, and development of such steels could result in useful economies in Magnox stations. For example, the introduction of a 36-ton steel, on conservative estimates, would lead to a reduction in the cost of electricity per unit sent out from a 250 MW station of 2½%. A stronger steel would of course introduce its own problems, but the incentive for its use is very great and it may be advantageous to consider a modification of the design code in the direction of Continental practice of designing on proof stress at working temperature. Sir Leonard concluded with the warning that if suitable steels do not become available the reactor designer will look elsewhere for his materials—an allusion to the potential competition of concrete.

TABLE 1. Details of U.K. Reactor Pressure Vessels

Reactor Site	No.	Contractors	Estimated highest steel temperature, °C	Type of steel	Plate thickness, in
Calder	4	Whessoe ..	340	Al grain refined	2
Chapelcross ..	4	Whessoe ..	340	Al grain refined	2
Berkeley	2	TNPG	340	Si killed ..	3
Bradwell	2	TNPG	370	Si killed ..	3 and 4
Hinkley Point ..	2	EE-BW-TW ..	340	Si killed ..	3
Hunterston ..	2	GEC	2.0	Al grain refined	2⅜ and 3
Trawsfynydd ..	2	APC	340	Si killed ..	3½
Dungeness	2	TNPG	250	Al grain refined	4 and 4½

HIGH TEMPERATURE PROPERTIES

In their paper on the "High Temperature Mechanical Properties of Steels used in Gas Cooled Reactor Pressure Vessels," Dr. J. M. Robertson and Mr. R. W. Nichols presented the results of a large number of creep and stress rupture tests carried out in several different laboratories over the past four years on carbon steels of the types that are being used in present-day reactor pressure vessels. Table 1 shows the materials involved. The demand for good creep resistance arises from the necessity for maintaining alignment of the standpipes and, although the creep specification will depend on a particular design, the limit has been set in a typical case to 0.2% strain in 200,000 hours. Except for the lower portions of the two Bradwell reactors where a low alloy steel has been used (temperatures are low and charging is from the top) all steel so far used in the U.K. for pressure vessels has been of the low carbon high manganese type (Table 2), the low carbon content being desirable from the point of view of welding, the high manganese content to increase rupture strength and a high manganese to carbon ratio to improve notch ductility. The differences in properties between the steels used can be attributed to the amount of aluminium added to the ladle. The effect of the aluminium is also related to the amount of nitrogen in the steel, the existence of free nitrogen tending to increase creep resistance and lower ductility (see later).

When undertaking experiments on the properties of reactor pressure vessel steels it is important to remember that the final vessel will be stress relieved at about 600°C for 36 hours. It was at one time assumed that this site stress relieving could be imitated by heating for three to five hours in the 600 to 650°C range followed by slow cooling, but later experimental work indicated that prolonged stress relieving changed the mechanical properties. As a result this treatment is now given to test specimens although in the results presented in the paper only the electric furnace steels have been so treated.

TABLE 2. Typical Compositions of U.K. P.V. Steels

Steel	Type	Specification of typical composition, %					Typical Al addition per ton of steel
		C	Si	Mn	S	P	
1	Silicon killed pressure vessel mild steel	0.16	0.17	1.3	0.03	0.03	None
2	Silicon killed pressure vessel mild steel	0.2	0.2	1.0	0.03	0.03	4-8 oz
3	Silicon killed pressure vessel mild steel	0.16	0.2	1.3	0.03	0.03	4-8 oz
4	Aluminium grain refined pressure vessel mild steel	0.16	0.13	1.3	0.03	0.03	2 lb

Creep Measurements

The different characteristics of the creep curves for the silicon killed steels and those with aluminium grain refinement show the superior low deformation creep strength of the former. The stress to cause 0.2% creep in 200,000 h at 400°C is shown to be 3.5 ton/in² for the aluminium refined and 7 ton/in² for the normalized silicon killed steel—B.S. 1500 permits 5.7 ton/in². These results are for open hearth steels; tests at higher temperatures are still being conducted. On the basis of Larson-Miller plots from the stress rupture tests, it is deduced that the stress to rupture in 200,000 h at 400°C is 12 ton/in² for the silicon killed steel and 8.4 ton/in² for aluminium grain refined steels. The stress permitted by B.S. 1500 at this temperature is 5.7 ton/in². Indications are that prolonged stress relieving which decreases the creep resistance may have less effect on the aluminium grain refined type. The general difference though between the two types of steels is taken as being due to the difference in the free nitrogen content.

The theoretical aspects of creep however were taken further by Dr. J. Glen and Mr. J. D. Murray in their paper on the "Metallurgical Aspects of Creep in Carbon and Low Alloy Steels," where they consider that the total creep resistance can be considered as made up of two components—inherent creep resistance which depends on the condition of the steel before testing and the latent creep resistance which depends on precipitation processes during actual testing. The previous comments on the influence of nitrogen were amplified and the scattering of creep properties is considered to be largely controlled by the active nitrogen content of the steel.

Graphitization

The probability of graphitization of steels in pressure vessels, particularly near the weldments, is a matter of some dispute and it was difficult to gauge how far the results presented in the paper by Dr. J. E. Harris, Mr. J. A. Whiteman and Prof. A. G. Quarrell on the "Sub-critical Graphitization of Steels and Related Alloys" could be extrapolated to actual pressure vessel conditions. In their paper they reported on the graphitization characteristics of a number of vacuum melted pure iron-carbon and iron-carbon-silicon alloys.

CORROSION

Contributions to the corrosion section of the Symposium were somewhat smaller than anticipated and the hoped for report from General Dynamics on some of the work conducted at their laboratories did not materialize. The three papers in the section were concerned with completely different aspects

of corrosion the first being the relevant problems of the Magnox and AGR reactors, the second being a review of experience in the United States on water reactors and the third dealing with the corrosion problems of the fast reactor, upon which we shall report next month.

Corrosion in CO_2

The first paper presented by Dr. T. Raine and Mr. C. E. Moore on the "Oxidation of Reactor Steels in Carbon Dioxide" contained a digest of the results obtained by the various members of the industrial consortia and of the AEA, who have conducted a collaborative investigation on the oxidation of steels by CO_2. From a corrosion point of view the reactor pressure vessel is not the most critical component in a plant in that the plate is heavy and the maximum temperature will not exceed 360°C. The gas ducting however must operate at temperatures in the range of 400/420°C, but as these also are of thick plate the behaviour is not as critical as in the heat exchangers which must operate at the same temperature and where the fins on the tubing will have a thickness of the order of $\frac{1}{16}$ in. In the heat exchangers it is possible for severe corrosion to cause a serious drop in the heat conductivity. The bellows are also a critical component as the temperatures will be the same as those of the ducts and the thickness of the walls is limited by the need for flexibility.

The coolant in the Magnox reactors is CO_2 at an operating pressure of 100-250 lb/in² circulating at a rate in excess of 10,000 ton/h and having a maximum temperature of over 400°C. It will also contain the reaction products with the graphite of the reactor core, and it is anticipated that there will be an equilibrium CO content of 1-1½% by weight; the water content will be between 10 and 20 ppm but on start-up the moisture content may be as high as 300-400 ppm.

From the results obtained it would appear that the heat exchanger steels are likely to be satisfactory up to 400°C in dry CO_2 or CO/CO_2 and at temperatures somewhat below 400°C in moist CO/CO_2. Under these conditions the corrosion rates are approximating to a parabolic law. Under more arduous conditions such as higher temperatures and moist gas conditions it is unlikely that these steels will be satisfactory as they show accelerating corrosive conditions. In dry CO/CO_2 at 450°C a sample of rimming steel was still corroding at a parabolic rate after 4,000 hours but it would not be safe to extrapolate beyond this time.

Extrapolating the worst results obtained with pressure steels at 400°C the expected loss of metal due to corrosion over 20 years will be of the order of 0.012 in. There will also be some corrosion on the outside of the vessel but a design tolerance of $\frac{1}{16}$ in is already included in current designs.

Corrosion tests on bellows steels have shown that at 400°C for 20 years penetration would be less than 0.005 in.

It would appear that in the pressure range 100-300 lb/in² the corrosion rate is independent of pressure.

On the other hand M. Colombie reported on work conducted in France which has shown corrosion to be directly dependent upon the partial pressure of CO_2.

There was considerable discussion on the break-away phenomenon where the corrosion departs from the parabolic law, and it was suggested that due to the differences in the behaviour of different surfaces a high surface finish was desirable in investigations. The proposal that this break-away phenomenon is associated with carburization of the material did not receive enthusiastic support. An additional cause mentioned was the presence of copper which has been demonstrated to have a serious detrimental effect on corrosion properties. There appeared also to be disagreement on the significance of moisture content, Dr. Stevens quoting recent results where the addition of moisture consistently increased the corrosion rate particularly with a rimming steel sample. The influence of cavity barriers was also mentioned but it became clear that the real mechanism of the break-away phenomenon is just not at present sufficiently understood for the true causes to be identified.

Corrosion in PWR and BWR Circuits

Mr. F. L. Laque opened his paper, "The Corrosion of Pressure Circuit Materials in Boiling and Pressurized Water Reactors," prepared in conjunction with Mr. M. A. Cordovi, with a review of the current major pressurized and boiling water projects in the United States and a comment on the materials that are available for their construction. The basic material has been the chromium-nickel austenitic stainless steel but the possibility of reducing costs has prompted several studies on the feasibility of substituting carbon and low alloy steels. So far the results of such explorations have not provided a sufficiently well-established basis for use of these materials in primary systems of water-cooled reactors. The most important deterrents have been the evidence of susceptibility to localized attack, hydrogen embrittlement, impairment of ductility to which must be added the problem of increased corrosion products. In Fig. 1 typical rates of attack by high purity water on stainless steel and carbon steel are reproduced. Comparisons are also given of the behaviour of type 347 stainless steel and Inconel, the improved stress corrosion resistance of this material being a chief attraction. The corrosion rates of the two are shown to be similar. Experiments on the corrosion resulting from galvanic couples with Inconel have shown that this effect is negligible.

Particularly in the BWR which uses live steam in the turbine, the extent of the contamination of the primary circuit by corrosion products is of particular importance and some results of tests made on type 347 stainless steel and Inconel are quoted. The variation of the figures with temperature are somewhat difficult to understand but also surprising is the effect of irradiation on corrosion rate, a number of tests indicating that for alloys which characteristically develop adherent oxide films the radiation corrosion is less than the corrosion in unirradiated conditions (see Fig. 2).

A great deal of work has also been undertaken to determine the corrosion to be expected in the turbines of BWRs where free oxygen must be allowed for. In brief, it is concluded that stainless steels are better than alloy steels which are better than carbon steels, but all materials exhibit pitting with high oxygen content while only carbon steels with low oxygen content.

With specific reference to the heat exchangers where the steel must be resistant to both the primary and the secondary water it has been found that all commercial grades of chromium-nickel stainless steel would exhibit cracking when the water contains chlorides and oxygen. It would appear that increasing the nickel content reduces susceptibility to stress corrosion cracking but no steel is immune when the nickel content is below 50%. It is possible

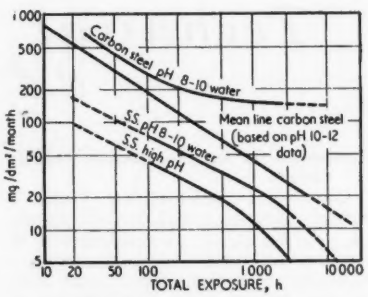


Fig. 1.—Descaled corrosion tests of AISI types 304 and 347 S.S. with carbon steel. At low pH, temp. is 500°F and at high pH (LiOH) temp. is 600°F. (La Que.)

that the nitrogen in the steel may be of critical significance.

Fabrication

In the fabrications sections of the Symposium details were given of the characteristics of raw materials and the fabrication techniques used at Bradwell and Dounreay in two papers, the first by Mr. W. C. Holliday and Mr. M. J. Noone on the "Metallurgical Problems Associated with Engineering and Fabrication of Gas Cooled Reactor Pressure Vessels," and the second by Mr. P. K. Richards on the "Fabrication of Reactor Vessel Heat Exchangers and Circuitry for the Dounreay Fast Reactor." These two papers form a valuable record of the actual work done on these two projects but no useful purpose would be served by attempting to summarize the wealth of detail given.

Radiation Damage

The effect of radiation on structural materials is a problem unique to the nuclear engineering field and the section devoted to this subject was opened by Dr. A. H. Cottrell who dealt with the "Theoretical Aspects of Radiation Damage and Brittle Fracture in Steel Pressure Vessels." Under the action of radiation, steels harden (see Fig. 3) and the tensile stress σ_y increases according to the empirical formula over the range of doses expected in reactor pressure vessels (10^{17}-10^{19} neutron/cm²) $\Delta\sigma_y = A(\phi t)^{1/3}$ where A is a constant, ϕ is the neutron flux and t the time of irradiation. Although there are signs that the hardening may saturate at really large doses in the core of the reactor the yield stress should rise by about 2 ton/in² in the first three hours and

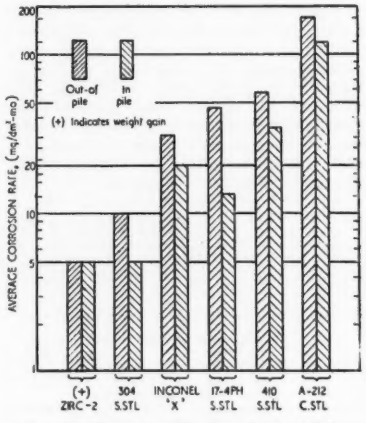


Fig. 2.—Corrosion of materials at 600°F, pH 8.9-9.5 in unirradiated and irradiated water. (La Que.)

20 ton/in² in the first six months. The mode of hardening can be separated into pinning and friction stresses but the hardening is due mainly to friction stress.

Loss of Ductility

The dominating effect of radiation, however, is the increase in the brittleness of the steel and the raising of the ductile-brittle transition temperature which can be assessed by the empiric relation: $\Delta T = 55(\phi t)^{1/3}$ where ϕt is the fast neutron dose in units of 10^{18}/cm². In this formula certain precautionary factors have been included and it implies that over a 10-year life the transition temperature of a power reactor vessel may rise by 50-100°C.

Considerable attention has been paid to the determining of the general question of safety of pressure vessels and how ductility should be measured, the mechanism of crack formation and its dependency upon design and operational characteristics and whether some relaxation could be allowed of the present strict ideas. The conclusion is drawn, however, that the chance of brittle fracture must become appreciable if the temperature of the vessel is allowed to fall 100°C below the crack propagation temperature and that the only really safe course is to maintain the operating temperature at all times above this temperature.

In their paper on "The Effect of Neutron Irradiation on the Ductile Transition Temperature of Steels and Its Relevance to Reactor Pressure Vessels" Dr. D. R. Harris, Mr. R. W. Nichols and Mr. C. Judge reproduced a vast quantity of Charpy notch tests on specimens irradiated to various levels. They pointed out that the change in transition temperature is markedly dependent upon the irradiation temperature, the increase being significantly more if the irradiation is given at 130°C than if at 50°C. At irradiation temperatures of around 300°C the extent of the damage is dependent on steel composition but in general is less than when the irradiation temperature is 130°C. The formula quoted by Dr. Cottrell represents substantially the worst conditions that are foreseen. If the change is compared in impact tests and slow bends tests it is found that the figures agree closely, at least according to the results on the aluminium treated low carbon pressure vessel steel irradiated at 130-156°C. The result of tests of various sized specimens indicated that size has an effect and that small specimens give somewhat optimistic figures. A comfort is the confirmation from specimens irradiated in Calder that the accelerated tests that are being conducted agree closely with those that have been experienced on the reactors themselves.

Annealing to Recover Ductility

In a report on American experience on the change in the transition temperature, Mr. J. R. Hawthorne in his paper "Studies of Irradiation Effects and Recovery of Notch Ductility of Pressure Vessel Steels" confirmed the comments of previous authors in terms of general magnitude and dependency on temperature but added to this comments on the effectiveness of heat treatment in restoring ductility. This is more effective on low temperature irradiated steels than on the higher temperature irradiations. Annealing a steel irradiated at 200°F, which had shown a shift in transition temperature of 185°F, for 80 hours resulted in a 75% recovery; further extension of this annealing time did not change the situation. A lower annealing temperature gave less recovery and a higher annealing temperature (700°F) resulted in 85% recovery. With the material irradiated at 575°F annealing at 650°F gave only 35% recovery and in another steel about 28% from 700°F annealing, the final transition shift being higher than for a cold irradiated specimen.

Papers were also given by Mr. L. E. Steele, of the U.S.A., on neutron dosimetry and by Mr. L. P. Trudeau, of Canada, on the influence of some metallurgical variables on embrittlement.

STEELS FOR THE FUTURE

The final session was devoted to a discussion of the developments that had taken place in this country as a result of the programme initiated by the Atomic Energy Authority for the production of high temperature and high tensile steels suitable for use in reactor pressure vessels. In view of the ability of the designer to limit temperatures the high tensile work would seem to be more important in reactor circuits. Four papers were presented "Progress and Development of Some Improved Steel for Gas Cooled Reactor Vessels," by Mr. I. M. MacKenzie of Colvilles; "The Development of a High Tensile Plate Steel for Use at Elevated Temperatures and Pressures," by Messrs. T. F. Pearson and M. Z. de Lippa of Consett Iron; "Steels for Nuclear Power Plant," by Dr. K. J. Irvine and Mr. J. D. Murray of United Steels; and "Niobium in Carbon and Low Alloy Weldable Steels," by Mr. J. E. Russell of English Steels. A brief summary of the properties of some steels is reproduced in Table 3. One objective has been to develop a steel in the 36 ton tensile range with at the same time good weldability and good creep (0.02% strain in 200,000 h) whilst at the same time retaining good ducitility properties. Details of the weldability of the steels was not given but the Conference was assured that this had been energetically pursued by the electrode manufacturers and plant fabricators and results were satisfactory.

Colvilles has concentrated upon improving the high temperature properties of Coltuf 28, an aluminium killed low carbon high manganese steel and made tests on additions of Mo up to 0.2%. A steel with 0.17% Mo appears to give 0.2% strain in 200,000 h at 7 ton/in² at about 420°C (as compared with 300°C for Coltuf 28) with good impact properties. For high tensile work Ducol W.30 has been developed which is a low carbon, Mn, Cr, Mo, V steel and tests have been made on plates of up to 4-in thickness. These show a UTS of 34 ton/in² after normalizing and tempering (equivalent to a vessel stress relief), the tempering accounting for a drop of 2 ton/in². Charpy tests on similarly treated material show the ductility to be marginally acceptable. Tests have also been made on quenched steel which shows a marked improvement in tensile but is scarcely suitable for large pressure vessels because of the stress relieving necessary.

Consett Iron have been studying a Mn-Ni-Mo-V steel and progressively modifying constituents to give the desired properties. Omitting the intermediate phases the percentage composition chosen for the final casts was 1.01 Mn, 1.58 Ni, 0.28 Mo, 0.11 V, 0.3 or 0.5 Cr. Thick plate tests show the steel with higher chromium content to have the better tensile properties, but poorer ductility and the 0.3% Cr seems more appropriate to reactor pressure vessels. It has been found that a double normalizing process is very beneficial and prolonged stress relieving does not lead to serious impairment of properties. The 0.3% Cr steel has a UTS of 36 ton/in² and shows an impact value of 35 ft lb at −10°C.

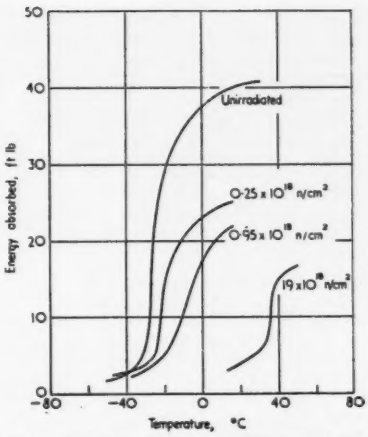


Fig. 3.—Effect of irradiation on Charpy v-notch impact specimens of En 2 steel (Cottrell).

The compositions are almost identical to those of Soudotenax 56 and 66, steels developed in Belgium on a parallel line and reported by M. H. Herbiet.

The United Steels contenders are Affreeze III in the 30 ton/in² UTS range (0.12% C, 1.36% Mn, 0.22% Mo, 0.07% V) which for a criterion of 0.1% creep in 100,000 h at a stress of 7 ton/in² would have a maximum surface temperature of 420°C. Tempering times of 600°C were limited in many of the tests to one hour but indications are that longer tempering times do not have a serious effect on this type of steel. The ductility properties are good. For higher temperature work (heat exchangers) Esshete CML (1% Cr-Mo) is recommended, and then Esshete CRM2, whilst for bellows and ducting there is interest in the Fortiweld series although no ductility figures were quoted.

The English Steels work on niobium additions is still in its early phases but interesting comments were made on the advantages of electric furnace steels both for the niobium steels and other steels—opinions not echoed by the O.H. proponents. The niobium steels would appear to hold promise for applications where the radiation dose is likely to be high needing a very low transition temp.

TABLE 3. Comparison of P.V. Steels (normalized or stress relieved conditions)

	Min. UTS (ton/in²)	Temp. for 35 ft/lb Charpy Value, °C	Robertson Crack Arrest Temp., °C		Creep Limit at ⅔ TS, °C
			Gradient	Isothermal	
Silicon killed pressure vessel steel	27	+20	+20	+40	350
A1-grain refined steel	27	−10	−20	+20	320
0.02% Nb steel	28	−40	−50	−15	~330
Grain refined + Mo	28	0	−10	—	>350
1.5 Ni, 0.3, Cr, Mo, V	36	−20	−20	− 6	~400
Ducol W.30	34	+ 0	−20	− 5	400
1.5 Ni, 0.5 Cr, Mo, V	40	+20	− 8	+45	400
Q.T. Cr, Mo, V	38	0	−40	0	420

ENEA

Basic Assumptions for Nuclear Power Estimates in Europe. Report published by the European Nuclear Energy Agency (OEEC).

This is a revision of a study published in 1958 entitled " Estimates of Nuclear Energy Production in Europe," in the light of the changes which have taken place in the past two years, not only in technology, but in revised national nuclear energy programmes. The report is detailed, but may be briefly summarized.

Installed Capacity

Based on the position in July, 1960, nuclear power stations under construction or irrevocably decided upon in OEEC countries totalled 4,500 MW(e), and projects of which a proportion could be expected to be completed by 1966 totalled another 3,500 MW(e). By 1966, then, it seems that the total installed capacity will be about 6,000 MW(e); representing only 5% of total European electricity production, and about 3% of total installed capacity.

Between 1967 and 1975, development will depend entirely on the ability to generate at competitive rates, and it is difficult to forecast beyond the next few years. It might, however, be envisaged that the total installed nuclear capacity might be 3%-9% of the total—i.e., 12,500-25,000 MW(e). There is no technical barrier to a figure of 35,000 MW(e), but it is considered unduly optimistic when considered in relation to the possible rate of expansion of industrial production generally.

Capital Costs

An extensive review of capital costs is made for a number of European and American reactors. The accompanying chart has been plotted from the figures given, and it is interesting to note that gas-cooled, PWR, and BWR all approximately follow the same pattern. Based on these figures, it is estimated that, up to 1966, costs will fall to $280 for gas-cooled, and $250 for light water-cooled per net kilowatt output.

Fuel Cycle Costs

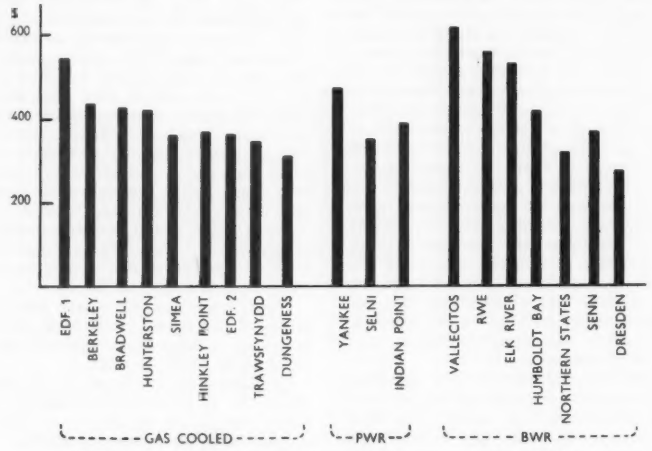
Once again, a comprehensive review is given of the fuel cycle costs. Based on reasonable extrapolation of existing parameters, it is estimated that for 1960-1966, the fuel cost of cycled uranium will be $175/kg for natural U, and $290/kg for enriched U.

Overall Expenditures

Total investment in nuclear generation, for stations under construction or irrevocably planned will, it is stated, amount to about $1,700M. Further projects expected to be in operation by then include 1050 MW(e) of natural U stations, and 450 MW(e) of enriched U. Based on the costs previously given, these would account for another $400M, giving a grand total of $2,100M. Total expenditure on fuel, taking into consideration the rate at which the various stations can be built and put into operation, will amount to $880M in the same period.

On the matter of payment in non-European currencies, the report states that it is reasonable to suppose that stations burning natural U will be wholly built by European companies. Purchases of uranium concentrates will amount to some $300M, of which 80%, or $240M, will be paid for in non-European currencies.

In the case of enriched U, however, a certain amount, say 25%, of capital costs will have to be paid in the form of dollars for machinery not available in Europe, for technical assistance and patent royalties. This would account for some $75M. It is advisable, also, to assume that all enriched fuel elements will be manufactured in the U.S. (although the conversion of hexafluoride and the fabrication of enriched fuel will undoubtedly be developed in Europe to some extent). This will account for a further $174M, leaving aside plutonium credit. The grand total of expenditure in non-European currency up to 1966 would be $490M.



Capital costs of nuclear power stations (Shippingport omitted) in $/kW.

LLOYD'S REGISTER

Survey of Pressure Components for Land-based Nuclear Installations: Provisional Requirements, 1960. Lloyd's Register of Shipping (Land Division).

At various times over the past 10 years, Lloyd's Register has made recommendations for the design, construction and testing of land-based nuclear components. Preliminary notes for the guidance of manufacturers were issued in 1957; these have been continuously reviewed as experience has been gained. It is now considered that the time has come to publish them in comprehensive form; it is intended to revise and supplement them as necessary.

Four classes of pressure vessels and components are recognized, according to the conditions of service:—

A. Pressure vessels and components intended to contain radioactive material and subject to neutron irradiation so that the physical properties of constructional materials are affected; or those which, from biological considerations, will be inaccessible for comprehensive examination and maintenance.

B. Vessels and components which are intended to contain radioactive materials, but are not subject to neutron irradiation, and can be examined or maintained on shut-down.

C. Vessels and components which neither contain radioactive materials nor are subject to neutron irradiation, and are available for examination and maintenance at any time.

D. Emergency containment vessels.

Detailed requirements are given, for both A and B, of general design, materials of construction, (plate, forgings, pipes, tubes and pressure shell attachments) methods of fabrication, (non-destructive examination, weld procedure and operator tests) and methods of testing after stress relieving.

No requirements are listed for category C, it being stated that manufacturers' proposals must be approved if applicable codes are not available.

Tentative requirements are given for category D vessels in substantially the same form as for A and B.

Throughout, extensive use is made of Lloyd's " Rules for Welded Pressure Vessels (Class 1) " and BS1500 (Class 1). It is pointed out, however, that consideration will be given to the use of national standards, in place of British standards, where these are reasonably equivalent.

Appendices cover requirements for primary circuit pressure relief systems, the ultrasonic examination of materials, plate-pressing procedure and production tests, weld procedure and operator tests. The two last-named appendices are complete with diagrams showing the stipulated positions of test pieces taken from a specimen plate; various types of test welds, and locations of impact test specimens cut from weld procedure test plates.

The publication, which runs to about 50 pp., is priced at £3 3s., and can be obtained from Lloyd's Register of Shipping (Land Division) at Norfolk House, Croydon.

MEETINGS

January 9.—The Institution of Electrical Engineers. (South Midland Supply and Utilization Group, at the College of Technology, Gosta Green, Birmingham.) **The Application of Irradiation in Industry.** M. C. Crowley-Milling, M.A.

January 11.—The Society of Instrument Technology. (Data Processing Section, London, at B.I.S.R.A., 6 p.m.) **The Tallimarker—An Information Handling Scheme for Processing Lines.** W. L. Marks, M.Sc., and D. J. Mayes.

January 18-19.—The Institution of Mechanical Engineers. **Symposium on Pressure Vessel Research Towards Better Design.**

January 24.—The Institution of Chemical Engineers. (London, at The Geological Society, Burlington House, W.1, 5.30 p.m.) **Lecture on Mass Transfer in Drops and Bubbles.** F. H. Garner.

January 27.—The Institution of Structural Engineers. (Midland Counties Branch, joint meeting with the Institute of Welding, Birmingham Branch, at the Byng Kenrick Suite, College of Advanced Technology, Gosta Green, Birmingham, 6.30 p.m.) **The Structural Engineer in the Field of Atomic Energy.** T. C. Waters, M.I.Struct.E., Chief Structural Engineer, U.K.AEA, Warrington.

January 31.—The Society of Instrument Technology. (London, at Manson House, 26 Portland Place, W.1, 6.30 p.m.) **Instrumentation, Past, Present and Future.** L. S. Yoxall.

Industrial Notes

To assist industrial companies in developing their export business the **Board of Trade** offer an all-embracing advisory, research and contact service. It provides information on every aspect of the export business—details of importers, outlets, agents, tariffs, samples, travelling insurance, trade fairs, publicity, statistics, publications, currencies and credits. It does all it can to make trading overseas as simple as possible. Further details of the service are given in a booklet, "British Government Services for Exporters," obtainable from the Board of Trade Export and Publicity Fairs Branch, Horse Guards Avenue, London, S.W.1. Included in its contents are the names, addresses, and telephone numbers of the headquarters branches and regional offices of the service.

Bio-Rad Laboratories, of Richmond, California, have introduced a lease/purchase plan for the supply of heavy water. Customers take delivery of the water, make use of it as they wish and then return it to the suppliers. They are then given credit based on the time they had the water and on the amount of dilution it has undergone.

British Overseas Fairs are organizing two specialized all-British trade fairs in Mexico City during 1961. The first, to be called British Equipment for Mexican Industry, will be open from September 29 to October 15 and the second, British Equipment for Mexican Development, open from November 3-19. Both fairs are sponsored by the Federation of British Industries and the Dollar Exports Council.

The Atomic Weapons Research Establishment displayed at the 1960 Corrosion and Metal Finishing Exhibition, London, methods of depositing protective metal coatings on uranium and other metals of nuclear energy interest. Studies have been carried out on the effects of ordinary atmosphere on these materials, for the purpose of overcoming storage problems and, in the case of uranium, for overcoming difficulties that could be encountered if uranium were to be used for other than nuclear applications.

Pye of Cambridge are to manufacture and sell AMF-designed research and training reactors in the United Kingdom and Eire. A licensing agreement has been concluded between the two companies. It is anticipated that the agreement will later be extended to cover British Commonwealth and certain Middle East countries.

The third international conference on computation is to be held in Belgrade, September 4-9. It is being organized by the **International Association for Analogue Computation** and the Yugoslav National Committee for Electronics Telecommunications, Automation and Nuclear Engineering.

British Furnaces and **Kepston** have formed a joint company, the Kepston Division of British Furnaces, for the production of electric resistance furnaces operable with controlled atmospheres.

International Scientific and Technical Service, of Pomona, California, have formed a new division to handle licensing arrangements and associated problems between U.S. firms and companies and overseas organizations. They are interested in handling accounts for the manufacturers of industrial products, chemicals and heavy machinery.

In the display of photographs selected by a committee of the Institute of British Photographers for the 1960 Industrial Photographic and Television Exhibition was this shot of one of the boilers at Bradwell being cleaned. It was submitted for exhibition by Head Wrightson and Co.

Denison Mines of Toronto have been awarded a contract for the delivery of 6.5 metric tons of uranium oxide by the Japan Atomic Fuel Corporation.

Atomics International, a division of North American Aviation, made sales worth $48,611,884 during the year ended September 30, 1960.

Southern Instruments, of Camberley, Surrey, have introduced a scheme which enables comprehensive electronic instrumentation to be acquired on a rental basis.

Bristol Aeroplane Plastics have introduced piping fabricated of low alkali gas glass filaments impregnated and bonded with thermosetting epoxide resin with which they hope to compete in the stainless and mild steels markets for pipe systems. Known as Epoch pipe it has high anti-corrosive properties as well as other properties that make it suitable for a wide range of applications.

M & C Nuclear Inc., a subsidiary of Texas Instruments, are to fabricate fuel elements for the Special Power Excursion Reactor Test-3 (SPERT-III) at the U.S. National Reactor Test Station, Idaho Falls. SPERT-III is one of a series of reactors designed and operated by Phillips Petroleum under contract for the U.S. AEC for the study of reactor behaviour under runaway conditions. M & C Nuclear supplied the original fuel loading for this reactor in 1958.

Branson Ultrasonic Division of Branson Instruments have introduced a compact ultrasonic system for production cleaning of small parts and sub-assemblies.

IBM United Kingdom have supplied the U.K. AEA with an IBM 7090 data processing system to replace the IBM 709 which has been in operation since July, 1959. The new machine is of similar design to the old one but is capable of operating from four to six times faster.

Nuclear-Chicago Corporation have made a bid for the control of the Texas Nuclear Corporation which under the plan would continue operation under its own name as a subsidiary or division of Nuclear-Chicago.

Matthew Hall and Co. have moved their headquarters to a new building in Tottenham Court Road, London. Called Matthew Hall House, it will accommodate an administrative and drawing-room staff of over 1,000.

J. Blakeborough and Sons are to supply 16 main gas duct isolating valves for the nuclear power station being built by The Nuclear Power Group at Dungeness. They are similar in design, though larger, to the valves supplied by the company for Berkeley and under order for Tokai Mura.

Johnson, Matthey and Co. have acquired a controlling interest in the Swedish precious metal company A/B Gösta Nyström, of Stockholm, which will now be known as A/B Nyström and Matthey.

Conforex is the name of a company recently formed for handling export problems such as consortium management, drafting of contracts, finance, E.C.G.D. cover, forwarding and shipping. Address is 2 Prince's Row, Buckingham Palace Road, London, S.W.1. Telephone: VICtoria 2681.

Brooke Tool Manufacturing Co. have formed their machine, jig and fixture division into a limited company, Brooke Tool Automation. It will be responsible for the design, manufacture and technical sales of unit heads, special purpose unit machines and jigs and fixtures.

G. and J. Weir have formed a subsidiary, Alley Compressors, to undertake business transferred to the Weir works from Alley & MacLellan (Polmadie).

Vokes, Ltd., have changed their name to Vokes Group, Ltd., and a subsidiary company, Vokes, Ltd., has been formed to take over the business and trading interests of the parent company.

Elliott, Automation A.G., the Swiss subsidiary of Elliott Automation have moved into new and larger offices at 32 Holbeinstrasse, Zurich. Telephone: 47 05 70.

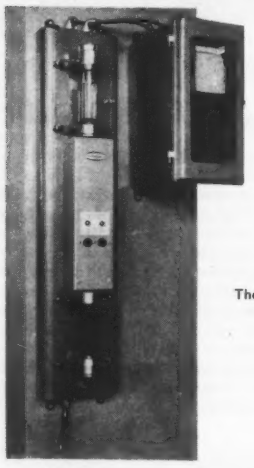
Northern Scientific Equipment have moved from Bolton, Lancs, to larger premises at Bridge Road Works, Lymington, Hants. Telephone: Lymington 3288.

The British Shipbuilding Research Association has moved to new offices at Prince Consort House, 27-29 Albert Embankment, London, S.E.11. Telephone number is Reliance 6481 and telegraphic address, Template, London, S.E.11.

The Index for the 1960 volume of *Nuclear Engineering* is to be included as an insert in this issue. Because it is not stitched in, it has been necessary to reduce the size of the page, but a limited number of uncut copies are available for those wishing to include them in bound volumes.

Water Monitor

Intended for the detection of beta-gamma activity in dissolved solids, the Elliott type ND.2151 monitor operates by passing about 20 litres/week through a bed of ion-exchange resin (Permutit Zeo-Karb 225) which surrounds a G-M tube operating a ratemeter and recorder.



The Elliott water monitor.

The flow of water to the sampling chamber is by an electrical timing mechanism in conjunction with a throttle valve. At the end of the complete week of operation the resin is removed by purging jets of water, and a fresh charge substituted. The average counting rate is continuously recorded and, if required, an additional unit can be provided for alarm actuation at a predetermined count rate. Sensitivity is such that full-scale deflection is obtained with 20 $\mu\mu$c of Sr-90.

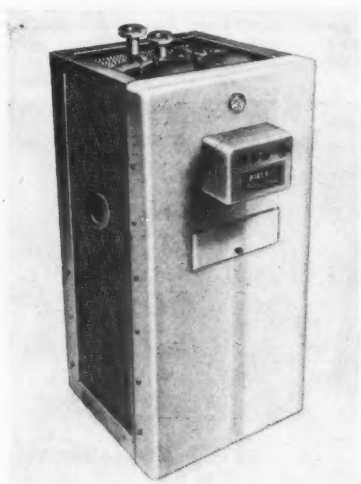
(Elliott Nucleonics, Ltd., Century Works, Conington Road, Lewisham, London, S.E.13.)

1399

New Gas Drying Units

New designs of Birlec continuous drying equipment for compressed air or gases, the BZ series Autosorbers, not only have advantages in price and performance, but show up to one-third saving in floor area. New methods of heating and cooling the adsorbers during reactivitation, and economy in purge gas are also claimed. The new designs are fully automatic dual-tower plants, capable of using any of the commercially available adsorbents, such as activitated alumina, molecular sieves, silica gel, Sorbead, etc., and typical sizes range from 25-500 ft³/min, at 100 p.s.i., or 5-100 ft³/min at atmospheric pressure. Standard units are fitted with electrical reactivitation; steam or flameproof electrical reactivitation can be fitted, if required.

A unit in the new Autosorber range.

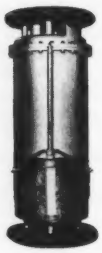


(AEI-Birlec, Ltd., Tyburn Road, Erdington, Birmingham, 24.)

1400

Lightweight X-ray Unit

The Baltospot G150, a new portable X-ray unit, achieves 150 kV performance with a weight of less than 52 lb. Cooling and insulation are by means of sulphur hexafluoride, with a built-in blower. The unit is available in two models, the type D with a 60° field, and the type P with a 360° field, the beam angle being 35° in either case, and the size of the focal spot 1.5 mm x 1.5 mm and 0.8 mm x 2.1 mm respectively. The head is offset, and a particular feature is the ability, in the panoramic version, to examine closed vessels of small diameter.



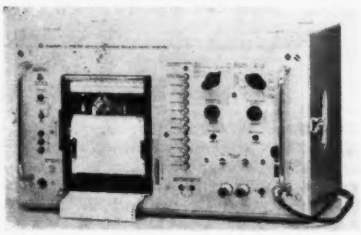
The Baltospot G150 portable X-ray unit.

(Pantak, Ltd., Vale Road, Windsor, Berks.)

1401

Multi-speed Chart Recorder

An improved model of the Rohde and Schwarz "Enograph" recorder is announced. A high-speed potentiometric instrument, it has facilities for linear and logarithmic indication, while its high input impedance (up to 10^{11} ohms) permits direct connection of diode rectifiers, photomultipliers, etc., where a variable can be expressed as a d.c. voltage. Ten chart speeds are provided with push-button selection; it is also possible to operate the chart from an external mechanism, if required. Four types of recording, ink, ball-point pen, electrode and stylus add to the instrument's versatility.



The Enograph chart recorder.

(Aveley Electric, Ltd., Ayron Road, Aveley Industrial Estate, South Ockendon, Essex.)

1402

BRIEFLY . . .

Frederick Jukes, Ltd., Royston Works, Garth Road, Lower Morden, Surrey, announce that they have acquired the sole manufacturing rights for Pontifex dust and fume control and recovery equipment designed by Bainbridge and Caplin, including automatic filter plant and wet washer units. **B200**

A new valve tester, the Model 45C, developed by **Taylor Electrical Instruments, Ltd.,** Montrose Avenue, Slough, Bucks, is said to be capable of testing more than 5,000 different types of British, American and Continental valves. It incorporates 21 valve holders and has test facilities for the earlier types of valve as well as the latest patterns. **B201**

Accurate and rapid identification of sound or corroded steel tubes and plates is possible with a new magnetic thickness gauge developed by **Solus-Schall, Ltd.,** County Building, Honeypot Lane, Stanmore, Middlesex. Weighing only 30 lb., the unit can be mains-operated or have built-in 12V batteries. Readings are direct, there being two ranges, to measure plates or tubes up to 8 mm thick. For those not wishing to purchase the equipment outright, an inspection service can be arranged. **B202**

New detergent cleaners for industrial and marine use have been introduced under the name Dasco Kleen by **D. A. Stuart Oil Co. (G.B.), Ltd.,** 11a Albemarle Street, London, W.1, an Amber Group subsidiary. The Dasco Kleen range includes 403, a general-purpose cleaner; 407, a germicidal mixture; 416, an economical degreaser; 415, a marine cleaner which readily emulsifies with salt water. **B203**

Nashton Mk. 1, claimed to be the first low-cost closed-circuit television equipment, has been announced by **Nash and Thompson, Ltd.,** Hook Rise, Tolworth, Surrey. Using a Vidicon pick-up tube, the camera is normally supplied with a 1 in f/1.9 lens, and has dimensions of 13½ in x 7¼ in x 7¼ in. A standard commercial TV receiver may be used as the monitor. For a single monitor, distances may be up to 2,500 ft, or 1,000 ft with 3-6 monitors, without booster amplifiers. **B204**

A new development in the lubrication of the flanges of overhead crane wheels is announced by **Acheson Colloids, Ltd.,** P.O. Box 12, Prince Rock, Plymouth, Devon. A stick of lubricant—a balanced blend of resin-bonded molybdenum disulphide and colloidal graphite—is contained in a spring-loaded holder, rubbing on the flange and leaving a dry film that will not attract dust. Although primarily intended for cranes, the system would appear to have possibilities in other situations where access is difficult or restricted. **B205**

The Plasinter Co., Ltd., Holyhead Road, Wednesbury, Staffs, have developed a new PVC finish material which can be applied to metal, wood, brick, concrete, asbestos, etc. Various grades are available, for application by brush or spray, and for stoving or air-drying. The material can be applied over existing paintwork where necessary. **B206**

A.E.I., Ltd., Radio and Electronic Components Division, 155 Charing Cross Road, London, W.C.2, announce a new Clix miniature wander plug and socket of dimensions suitable for printed circuit work. **B207**

Two new Muller (Hamburg) portable X-ray units for 140 and 200 kV are announced by **Research and Control Instruments, Ltd.,** Instrument House, 207 King's Cross Road, London, W.C.1. Consistent radiographs are obtained by pre-selection of tube voltage and exposure time, both being automatically controlled and capable of exact repetition by a single push-button. **B208**

Bardic, Ltd., William Street, Northam, Southampton, are marketing a new 2-cell pocket torch which will stand up to the same treatment as the previous 3-cell type. This has withstood salt water immersion, 48 hours at −20°F, and being dropped 15 ft on to concrete. A model carrying a safety certificate for use in hydrogen, ethylene and pentane is also available. **B209**

A new red dye-line printing paper is being marketed by **Lawes Rabjohns, Ltd.,** Abbey House, Victoria Street, London, S.W.1. It is particularly suitable for fast experimental work, where clear prints are required from weak pencil tracings. The prints can also be used as masters. **B210**

Electrolytic Time Meter

Early forms of electricity meters are recalled in new guise in the form of an indicator for elapsed time known as the Selachron, which has no moving parts, and requires no switches or relays. A small replaceable electrochemical

The Selachron electrolytic time meter.

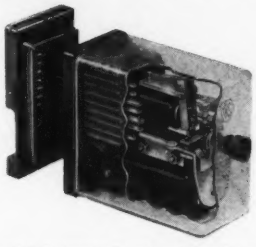
cell, containing anode, cathode, and electrolyte forms the basis of the indicator; a calibrated scale by the side of the anode allows the reduction in length to be read off in terms of hours of operation. Manufactured for use on a.c. or d.c., from 6-300 V, the Selachron is only 1 in diameter × 2¾ in long, and weighs less than 2 oz.

(Industrial Instruments, Ltd., 9 Paved Court, Richmond, Surrey.)

1403

New Latching Relay

A recent addition to the Clifford and Snell range of plug-in relays is a latching relay with electrical release, operating as a bistable



The C & S electrically released latching relay.

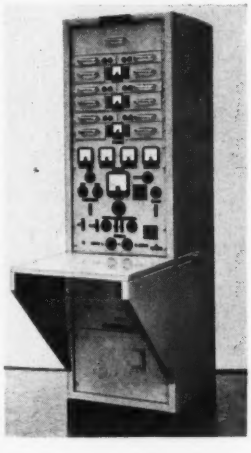
or memory relay. A short pulse to the operating coil causes a change in the contacts (up to three changeover or five pairs of make or break) persisting till the release coil is pulsed. All materials are non-inflammable and suitable for tropical conditions. A new manual-reset latching relay has also been introduced.

(D. Robinson and Co., Ltd., 717 London Road, Hounslow, Middlesex.)

1404

System Simulator

The Feedback SS 110 system simulator has been developed for laboratory use and for educational purposes where precision analogue computing equipment would not be justified on account of the cost. It has many uses in industrial development and research, but is particularly intended for teaching the basic principles of analogue and simulator techniques, and can simulate a wide range of systems described by linear differential equations of the second order. It is suitable either for real time or repetitive scales up to system frequency. It consists of three two-in-one d.c. amplifiers, eight interconnecting units, a square-wave unit, and a stabilized d.c. power supply, and there is a wide choice of component values of 1% accuracy.



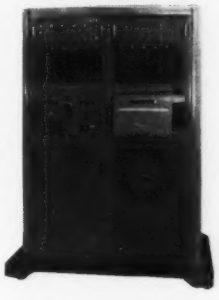
The Feedback SS 110 system simulator.

(Feedback, Ltd. (associate of Servomex Controls, Ltd.), Crowborough, Sussex.)

1405

Automatic Graph Plotter

The Electroplotter Model QL has been developed to enable a "quick look" at



The Benson-Lehner Electroplotter.

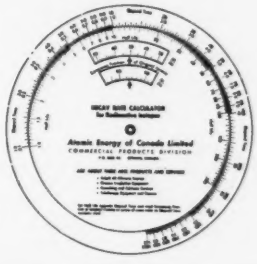
data recorded digitally on punched cards or tape at equal intervals of time (or other independent variables). Scales and offsets are preset by push-buttons; the machine then plots up to 12 variables, at up to 2,400 points/min.

(Benson-Lehner (G.B.), Ltd., West Quay Road, Southampton.)

1406

Isotope Calculator

Users—and potential users—of isotopes will be interested in a decay rate calculator devised by the Canadian atomic energy authorities, with which it is possible to esti-



The AECL decay-rate calculator for isotopes.

mate rapidly the proportion of energy remaining in an isotope of known half-life at the end of any period of lapsed time.

The calculator has been issued simultaneously with a handbook of general information on isotopes and their application. Either or both may be obtained by *bona fide* applicants; a nominal charge is made for quantities.

(Atomic Energy of Canada Limited, Commercial Products Division, P.O. Box 90, Ottawa.)

1407

CATALOGUES

Leaflets on filtration plant and deaeration equipment have been received from **Head Wrightson and Co., Ltd.,** Ship House, 20 Buckingham Gate, London, S.W.1.

Jeltek protective and waterproof clothing for site work is described in leaflets received from **Jeltek, Ltd.,** Green Lane, Hounslow, Middx.

Hiduminium 400, the aluminium-boron carbide dispersion described in *Nuclear Engineering* for June, 1960 (p. 283), is the subject of a booklet received from **High Duty Alloys, Ltd.,** Slough, Bucks.

Henry Wiggin and Co., Ltd., Thames House, Millbank, London, S.W.1, have issued a new booklet describing Inco-Weld "A" for the welding of dissimilar metals.

Negretti and Zambra, Ltd., 122 Regent Street, London, W.1, have issued a German version of their mercury-in-steel recording thermometer pamphlet, and a new Indian version of their general catalogue.

A brochure on their non-destructive testing service has been received from **Solus-Schall, Ltd.,** County Building, Honeypot Lane, Stanmore, Middx.

Connectors for ribbon feeders (flat, flexible, insulated conductors) are described in a leaflet issued by **Belling and Lee, Ltd.,** Great Cambridge Road, Enfield, Middx.

Fabrication of beryllium tubes and sections is the subject of a brochure issued by the **Chesterfield Tube Co., Ltd.,** Chesterfield, a subsidiary of Tube Investments, Ltd.

Panellit, Ltd., Elstree Way, Borehamwood, Herts, have issued a catalogue of annunciator systems.

"Airbrasive" units, for rapid and accurate cutting of very hard materials with a thin high-speed stream of abrasive particles, are described in a bulletin from the **S. S. White Industrial Division,** 10 East 40th Street, New York 16, New York.

A brochure commemorating their jubilee has been issued by the **Star Aluminium Co., Ltd.,** Wolverhampton, Staffs.

Opportunities for chemists and chemical engineers are described in a booklet issued by **Albright and Wilson, Ltd.,** 1 Knightsbridge Green, London, S.W.1.

Volumetric and lamp-blown glassware are described in a 54-page priced catalogue recently issued by **Griffin and George, Ltd.,** Ealing Road, Alperton, Middx.

A booklet on Eurochemic, the European company for the processing of irradiated fuels, has been issued by the **European Nuclear Energy Agency,** 38 Boulevard Suchet, Paris 16e.

"T.I. Extended Surface Tube" is the title of a publication received from **Tube Investments, Ltd.,** The Adelphi, London, W.C.2, which describes the various forms of extended circuits manufactured by their subsidiary companies, Tube Products, Ltd., Richards and Ross, Ltd., Howell and Co., Ltd., Accles and Pollock, Ltd., and Tubes, Ltd.

Alfred Imhof Ltd., 112-116 New Oxford Street, London, W.C.1, have issued a catalogue of electronic racks, cabinets, consoles, and accessories, in 10 languages (English, French, German, Spanish, Italian, Dutch, Portuguese, Norwegian, Swedish and Danish).

Rotary vacuum filters are described in a new publication from **Davey, Paxman and Co., Ltd.,** Colchester, Essex.

Patents Reviewed

These abstracts have been made from British Patent Specifications, complete copies of which can be obtained from the Patent Office, 25 Southampton Street, London, W.C.2, at 3s. 6d. each (including postage).

B.P. 836,043. Support structures for nuclear reactor fuel elements. J. Tatlock, J. A. F. Glass. To: U.K. Atomic Energy Authority.

Part-way along their length the fuel elements in a fast reactor core are supported by a nest of tubes in face contact with one another on hexagonal sections in the tubes within sections of greater area. The centres of the tubes correspond with the corners of a symmetrical hexagonal lattice. Tubes of greater length alternate with tubes of lesser length fixed longitudinally by their intersecting positions.

B.P. 836,047. Fission product separator for liquid reactor systems. G. V. Hough. To: Plessey Co., Ltd.

Refers to means for indicating that a particular fuel element has failed. It is necessary for this purpose to scrub fission products from a given channel selected according to some system. A simple arrangement of individual channel selecting valves within the scrubbing unit permits the delivery of liquid directly into the gas inside the scrubbing chamber.

B.P. 836,228. Winches for lifting radioactive articles. H. C. Knights. To: U.K. Atomic Energy Authority.

A winch drum is surrounded by a biological shield split below the axis of the drum into an upper and a lower part. When the parts are separated a segment of the drum is exposed. The rotating means is housed in the upper part.

B.P. 836,490. Time of flight measurement neutron spectrometer. To: Commissariat à l'Energie Atomique (France).

For use in connection with nuclear reactors. Neutrons penetrate periodically into a passage provided in a rotor which constitutes the neutron selecting element. This rotor is driven by a hydraulic turbine actuated by a lubricating fluid identical to that used for lubricating the other part of the apparatus so that a single hydraulic circuit can be used, leaks being recovered prior to recycling in the casing of the apparatus.

B.P. 836,562. Nuclear reactor fuel element handling plant. R. A. Taylor. To: Babcock and Wilcox, Ltd.

Fuel elements withdrawn from the reactor are passed through a (helical) chute so sloped that acceleration of the elements is small or prevented by friction. Discharging liquid (water jets) assists the movement of the elements so that they reach the discharge level of the chute with a low velocity.

B.P. 836,563. Material handling plant, more particularly plant for handling nuclear reactor fuel elements. R. A. Taylor. To: Babcock and Wilcox, Ltd.

Elements discharged from a chute as described in B.P. 836,562, are transferred from the chute by a skip hoist, the carriage of which moves the containers from the loading station (in a sealed chamber where the discharge end of the chute is located) to an unloading station within the storage pond, and accessible from above. A control point remote from the loading station controls the loading of an empty container and gives automatic indication of the presence or absence of a container at the loading station to avoid discharge when no container is available.

B.P. 837,769. Nuclear reactor. To U.S. Atomic Energy Commission (U.S.A.).

A core having fissile fuel is enclosed in a fertile blanket with a movable portion to control the amount of leakage of neutrons. This core comprises a series of thermo-electric assemblies in a housing, each including fissionable fuel as active element to form a hot junction of a thermocouple. The assemblies are so arranged that an electrically conductive coolant can be introduced between them to form cold junctions so as to develop an electromotive force across the entire series due to fission heat generated in the fuel.

B.P. 837,773. Portable boring machine. W. Lord, H. S. Royce. To: U.K. Atomic Energy Authority.

This machine is specially adapted to drill holes in convex surfaces.

B.P. 837,785. Nuclear reactor. A. J. Taylor. To: Babcock and Wilcox, Ltd.

The fuel channels are upright and the body of moderating material is constrained at its sides by a circumferentially sectionalized wall around the body. These sections are coupled together by compensating means so that differential thermal expansion and contraction of the body does not affect the constraining action of the wall portions supporting the body.

B.P. 837,786. Power plant. A. J. Taylor. To: Babcock and Wilcox, Ltd.

Valve means are provided in the circuit for the flow of fluid through a ship's reactor and the heat utilizing means in order to isolate a part of the circuit. The control may be exercised by fluid pressure responsive means acting on the occurrence of leakage through mechanical damage.

B.P. 837,848. Homogeneous nuclear reactor. To: Esso Research and Eng. Co. (U.S.A.).

The reactor shell is provided with a number of ports. A plurality of conduits is disposed external to the shell each communicating with two ports providing several external circuits, in each of which is arranged at least one heat exchanger. The sum of the cross-sectional areas of the ports is at least twice the maximum interior diameter of the shell. The streams of fuel may further be maintained in separate conduits in the core, provided the fuel is present in critical mass concentration. The arrangement has the advantages of (1) extremely effective cooling; (2) extraordinary high radiation fluxes. It may be used for thermal, intermediate, and fast reactors.

B.P. 837,853. Nuclear reactor fuel elements. J. Williams, W. Munro. To: U.K. Atomic Energy Authority.

A uranium rod is inserted in a blind hole in a beryllium body which is closed by a beryllium plug. The composite billet is then extruded at 420-450°C under a pressure of 40-50 ton/in.^{2}

B.P. 837,878. Manufacture of nuclear fuel elements. To: Siemens-Schuckertwerke A.G. (Germany).

An empty sheath is placed in a compression mould and filled with layers of powdered fuel. The layers are compressed by a ram and the compacted layers subjected to a solidifying operation (cold or hot pressing or sintering under pressure). The sheath is then closed. To prevent decrease in volume by sintering, a fuel filler core may be placed in the sheath before introducing the fuel.

B.P.837,920. Pipe joints. R. S. Challender, R. P. Kinsey. To: U.K. Atomic Energy Authority.

Refers to welded "T"-joints in positions difficult of access. The weld is made by access through a hole in the opposed wall later closed by an externally welded closure cap.

B.P. 837,930. Method of making fuel elements. To: U.S. Atomic Energy Commission (U.S.A.).

A cladding frame is formed of separate parts around a metal core (uranium) which is to be covered by the material of the frame (zirconium). The assembly is sealed, the interior purged with an inert gas, then evacuated and the assembly placed in a jacket of a third material, heated and rolled to bond it together as a unit.

B.P. 836,843. Nuclear reactor. To: United States Atomic Energy Commission (U.S.A.).

In a power reactor the time for the replacement of fuel elements must be minimized. The fuel elements are strands of fissionable material (uranium) in zirconium packets with helical ribs. They are arranged in concentric circles in a vertical pressure vessel and suspended between upper and lower dished heads. The dished heads are provided with aligned openings from which sleeves extend outwardly. When replacement is necessary the reactor is shut down, the heavy water drained from the vessel and the sleeve closures are removed by remotely controlled wrenches. On rotating apertured discs the strands of fuel material are released and drop down into a canal. Then new fuel sections are inserted from the other end in a similar manner and the sleeves sealed again. In the described reactor 114 fuel assemblies are used with 13 control rods of cadmium in zirconium jackets. Heavy water acts as coolant and moderator.

B.P. 837,443. Rotary selector valves. E. Long, J. R. Gurner. To: Plessey Co., Ltd., U.K. Atomic Energy Authority.

The valve is intended for an arrangement for taking gas samples under pressure selectively from a series of pipes to be coupled to a single port. Previous arrangements presented a great number of leakage possibilities, while the new valve possesses the feature that dilution of the gas being sampled can only occur at two places. Maintenance is possible without disconnecting any pipe connections. The valve has a detachable head, the ports being arranged in the body in two banks. A rotary connector arm at one end of the driving shaft passes through the head, and has at its other end duplicated mouths sealable against the body and coupling with the ports one at a time for fluid to flow through the mouths alternately as the arm rotates.